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Docket No. 50-254

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

Gentlemen:

By letter dated February 28, 1974, and supplemented by your letter dated January 28, 1974, you requested authorization to operate the Quad-Cities Unit 1 with 7 x 7 and 8 x 8 reload fuel assemblies for the first refueling of the Unit 1 core. In addition, your letter dated March 18, 1974, submitted proposed changes to the Appendix A, Technical Specifications of DPR-29, associated with the first refueling of the Quad-Cities Unit 1 core.

We have reviewed the above submittals and have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation of Quad-Cities Unit 1 with the first reload core, as described in the above submittals, and implementation of the proposed changes to the Technical Specifications (Appendix A) to DPR-29.

Accordingly, Amendment No. 10 to the Quad-Cities Unit 1 Facility Operating License No. DPR-29 is enclosed revising the Technical Specifications thereto to authorize operation of the Quad-Cities Unit 1 facility with the first reload core. Our related Safety Evaluation is enclosed.

A copy of a notice which is being forwarded to the Office of the Federal Register for publication relating to this action also is enclosed for your information.

Sincerely,

/S/

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

CHebron
NDube
MJinks (4)
BScharf (15)
RMDiggs
SKari
SVarga
RTedesco
GLainas
CLong
PCollins

Called CE Co - Abel on 6/6/74: advised of this concurrence. JIR 6/6/74

Enclosures and cc: See next page

OFFICE	L:ORB #2	L:ORB #2	L:ORB #2	OGC	L:OR	APPV
SURNAME	RMDiggs:rwg	JIRiesland	DLZiemann	R. Kinsey	KRGoller KRG	
DATE	5/23/74	5/23/74	5/31/74	6/4/74	6/5/74	

Form AEC-318 (Rev. 9-53) AECM 0240 GPO C43 16 81465-1 520-284

Enclosures:
1. Amendment No. 10 to DPR-29
2. Safety Evaluation
3. Federal Register Notice

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
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Moline Public Library
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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 & cy of CE's
ltr dtd 3/18/74:
Mr. Gary Williams
Federal Activities Branch
Environmental Protection Agency
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Mr. Leroy Stratton
Bureau of Radiological Health
Illinois Department of Public Health
Springfield, Illinois 62706

Anthony Z. Roisman, Esquire
Berlin, Roisman and Kessler
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Washington, D. C. 20036

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-254

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated February 28, 1974, including the supplements dated January 28, 1974 and February 7, 1974 (filed prior to the basic application), and subsequent supplement dated March 18, 1974, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

2. Accordingly, Paragraph 3.B of Facility License No. DPR-29 is hereby amended to read as follows:

   "(3) Technical Specifications

   The Technical Specifications contained in Appendices A and B, attached to Facility Operating License No. DPR-29 are revised as indicated in the attachment to this license amendment.

OFFICE ➤						
SURNAME ➤						
DATE ➤						

The Technical Specifications, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised."

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

FOR THE ATOMIC ENERGY COMMISSION

/s/

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

Attachment:
Change No. 19 to Appendix A
Technical Specifications

Date of Issuance: JUN 05 1974

OFFICE ➤						
SURNAME ➤						
DATE ➤						

ATTACHMENT TO LICENSE AMENDMENT NO. 10

CHANGE NO. 19 TO TECHNICAL SPECIFICATIONS (APPENDIX A)

FACILITY OPERATING LICENSE NO. DPR-29

The attached pages supersede pages bearing the same number, except as otherwise indicated. The revised pages have marginal lines indicating where the changes appear.

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Page 109B--Replacing existing pages 109B and 109C issued 5/14/74.
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1.1 Safety Limit Bases (cont'd)

psig or 5% flow. In general, Specification 1.1.B will only be applicable during startup, hot standby, or shutdown of the plant. A review of all the applicable low pressure and low flow data[1,2] has shown the lowest data point for transition boiling to have a heat flux of 144,000 Btu/hr/ft^2. To assure applicability to the Quad-Cities fuel geometry and provide some margin, a factor of 1/2 was used to obtain the critical heat flux; i.e., critical heat flux was assumed to occur for these conditions at 72,000 Btu/hr/ft^2. Assuming a peaking
3 factor of 3.06 this is equivalent to a core average power of 460 MW(t) (18% of rated). This value is applicable to ambient pressure and no flow conditions. For any greater pressure or flow conditions, there is increased margin.

During transient operation the heat flux (thermal power-to-water) would lag behind the neutron flux due to the inherent heat transfer time constant of the fuel which is 8-9 seconds. Also, the limiting safety system scram settings are at values which will not allow the reactor to be operated above the safety limit during normal operation or during other plant operating situations which have been analyzed in detail[3]. In addition, control rod scrams are such that for normal operating transients the neutron flux transient is terminated before a significant increase in surface heat flux occurs. Scram times of each control rod are checked each refueling outage to assure the insertion times are adequate. Exceeding a neutron flux scram setting and a failure of the control rods to reduce flux to less than the scram setting within 1.5 seconds does not necessarily imply that fuel is damaged; however, for this specification a safety limit violation will be assumed any time a neutron flux scram setting is exceeded for longer than 1.5 seconds.

If the scram occurs such that the neutron flux dwell time above the limiting safety system setting is less than 1.7 seconds, the safety limit will not be exceeded for normal turbine or generator trips, which are the most severe normal operating transients expected. These analyses show that even if the bypass system fails to operate, the design limit of MCHFR = 1.0 is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided with the Quad-Cities units has a sequence annunciation program which will indicate the sequence in which scrams occur such as neutron flux, pressure, etc. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient. Thus, computer information normally will be available for analyzing scrams; however, if the computer information should not be available for any scram analysis, Specification 1.1.C.2 will be relied on to determine if a safety limit has been violated.

During periods when the reactor is shut down, consideration must also be given to water level

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(1) E.Janssen, "Multi-Rod Burnout at Low Pressure," ASME Paper 62-HT-26, August 1962.

(2) K.M.Becker, "Burnout Conditions for Flow of of Boiling Water in Vertical Rod Clusters," AE-74 (Stockholm, Sweden), May 1962.

(3) SAR, Section 4.4.3 for turbine trip and load reject transients, Section 4.3.3 for flow control full coupling demand transient, and Section 11.3.3 for maximum feedwater flow transient. "Quad Cities 1 Nuclear Power
19 Station First Reload License Submittal," Section 6.2.4, Feb. 1974. Dresden Station Special Report No. 29 Supplement B."

subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. To accomplish their intended function, the capacity of each emergency core cooling system component was established based on the reactor low water level scram setpoint. To lower the setpoint of the low water level scram would increase the capacity requirement for each of the ECCS components. Thus, the reactor vessel low water level scram was set low enough to permit margin for operation, yet will not be set lower because of ECCS capacity requirements.

The design of the ECCS components to meet the above criteria was dependent on three previously set parameters: the maximum break size, the low water level scram setpoint and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could lead to a loss of effective core cooling. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

E. Turbine Stop Valve Scram - The turbine stop valve scram like the load rejection scram anticipates the pressure, neutron flux, and heat flux increase caused by the rapid closure of the turbine stop valves and failure of the bypass. With a scram setting at 10% of valve closure the resultant increase in surface heat flux is the same as for the load rejection and thus adequate margin exists.

No perceptable change in MCHFR occurs during
the transient[1]. Ref. Section 11.2.3 SAR and
19 Dresden Station Special Report No. 29 Supplement B.

F. Turbine Control Valve Fast Closure Scram -
The turbine control valve fast closure scram
3 is provided to anticipate the rapid increase
in pressure and neutron flux resulting from
fast closure of the turbine control valves
due to a load rejection and subsequent failure
of the bypass; i.e., it prevents MCHFR from
becoming less than 1.0 for this transient.
For the load rejection from 100% power, the
heat flux increases to only 106.5% of its
rated power value which results in only a
small decrease in MCHFR[1]. Ref. Section 4.4.3
SAR and Dresden Station Special Report No. 29
19 Supplement B.

G. Reactor Coolant Low Pressure Initiates Main
Steam Isolation Valve Closure - The low pressure
isolation at 850 psig was provided to give
protection against fast reactor depressurization
and the resulting rapid cooldown of the vessel.
4 Advantage was taken of the scram feature
which occurs in the run mode when the main
steam line isolation valves are closed to provide for reactor shutdown so that operation at
pressures lower than those specified in the
thermal hydraulic safety limit does not occur,
although operation at a pressure lower than
850 psig would not necessarily constitute an
unsafe condition.

H. Main Steam Line Isolation to Valve Closure Scram - The low pressure isolation of the main steam lines at 850 psig was provided

(1) "Quad Cities 1 Nuclear Power Station First
19 Reload License Submittal," Section 6.2.4,
February 1974.

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
3 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
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[4] limit the pressure to approximately 1100 psig.
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
19 13 (5,6,7 & 8) which is at least 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. "Quad Cities 1 Nuclear Power Station First Reload License Submittal,"
19 Section 6.2.4.2, February 1974.
(5) SAR Section 4.4.3. "Quad Cities 1 Nuclear Power Station First Reload License Submittal," Section 6.2.4.2, February 1974.
(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.
(8) Dresden Station Special Report No. 29 Supplement B.

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
19 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 bottom 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.

Unit 2

The pressure at the bottom of the vessel is about 1240 psig when the first safety valve opens and about 1280 psig when the last valve opens. Both values are clearly within the code requirements.

Vessel dome pressure reaches about 1305 psig with the peak at the bottom of the vessel near 1330 psig. Therefore, the pressure scram and safety valve actuation provide adequate margin below the peak allowable vessel pressure of 1375 psig.

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

3.2 Limiting Condition for Operations Bases (cont'd)

The APRM rod block function which is set at 12% of rated power is functional in the refuel and Startup/Hot Standby mode. This control rod block provides the same type of protection in the Refuel and Startup/Hot Standby mode as the APRM flow biased rod block does in the run mode; i.e., it prevents MCHFR from decreasing below 1.0 during control rod withdrawals and prevents control rod withdrawal before a scram is reached.

The RBM rod block function provides local protection of the core, i.e., the prevention of critical heat flux in a local region of the core, for a single rod withdrawal error from a limiting control rod pattern. The trip point is flow biased. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked when
19 Unit 1 MCHFR is $\approx$1.08 and Unit 2 MCHFR is $\approx$1.6 thus allowing adequate margin[1]. Below $\approx$ 70% power the worst case withdrawal of a single control rod results in a MCHFR $>$1.0 without rod block action, thus, below this level it is not required.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCHFR approaches 1.0. Ref. SAR Section 7.4.4.3.

A downscale indication on an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 3/125 of full scale.

The SRM rod block with $\leq$ 100 cps and the detector not fully inserted assures that the SRM's are not withdrawn from the core prior to commencing rod withdrawal for startup. The scram discharge volume
4 high water level rod block provides annunciation for operator action. The alarm setpoint has been selected to provide adequate time to allow determination of the cause of level increase and corrective action prior to automatic scram initiation.

For effective emergency core cooling for small pipe breaks, the HPCI system must function, since reactor pressure does not decrease rapidly enough to allow either core spray or LPCI to operate in time. The automatic pressure relief function is provided as a backup to the HPCI in the event the HPCI does not operate. The arrangement of the tripping contacts is such as to provide this function when necessary and minimize spurious operation. The trip settings given in the specification are adequate to assure the above criteria are met. Ref. SAR Section 6.2.6.3. The specification preserves the effectiveness of the system during periods of maintenance, testing, or calibration, and also minimizes the risk of inadvertent operation; i.e., only one instrument channel out of service.

Two air ejector off-gas monitors are provided and, when their trip point is reached, cause an isolation of the air ejector off-gas line. Isolation is initiated when both instruments reach their high

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19 (1) "Quad-Cities Unit 1 Nuclear Power Station First Reload License Submittal," Section 6.2.3, Feb. 1974

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
4 imperfect mixing. A boron concentration of 600 ppm in the reactor core is required to
19 bring the reactor from full power to a **Unit 1 3% Δk and Unit 2 5% Δk** subcritical condition considering the hot to cold reactivity swing, xenon poisoning and
4 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

3. If Specification 3.5.H.1 and 2 cannot be met, reactor startup
4 shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

I. Average Planar LHGR

During steady state power operation, the average linear heat generation rate (LHGR) of all the rods in any fuel assembly, as a function of
19 average planar exposure, at any axial location, shall not exceed the maximum average planar LHGR shown in Figure 3.5.1.

4.5 SURVEILLANCE REQUIREMENT

c. The RHR service water pump and diesel generator cooling water pump bed plate drains shall be checked during each operating cycle by assuring that water can be run through the drain lines and actuating the air operated valves by operation of the following sensors:

   i. loss of air
4 ii. equipment drain sump high level
   iii. vault high level

d. The condenser pit 5 foot trip circuits for each channel shall be checked once a month. A logic system functional test shall be performed during each refueling outage.

I. Average Planar LHGR

19 Daily during reactor power operation, the average planar LHGR shall be checked.

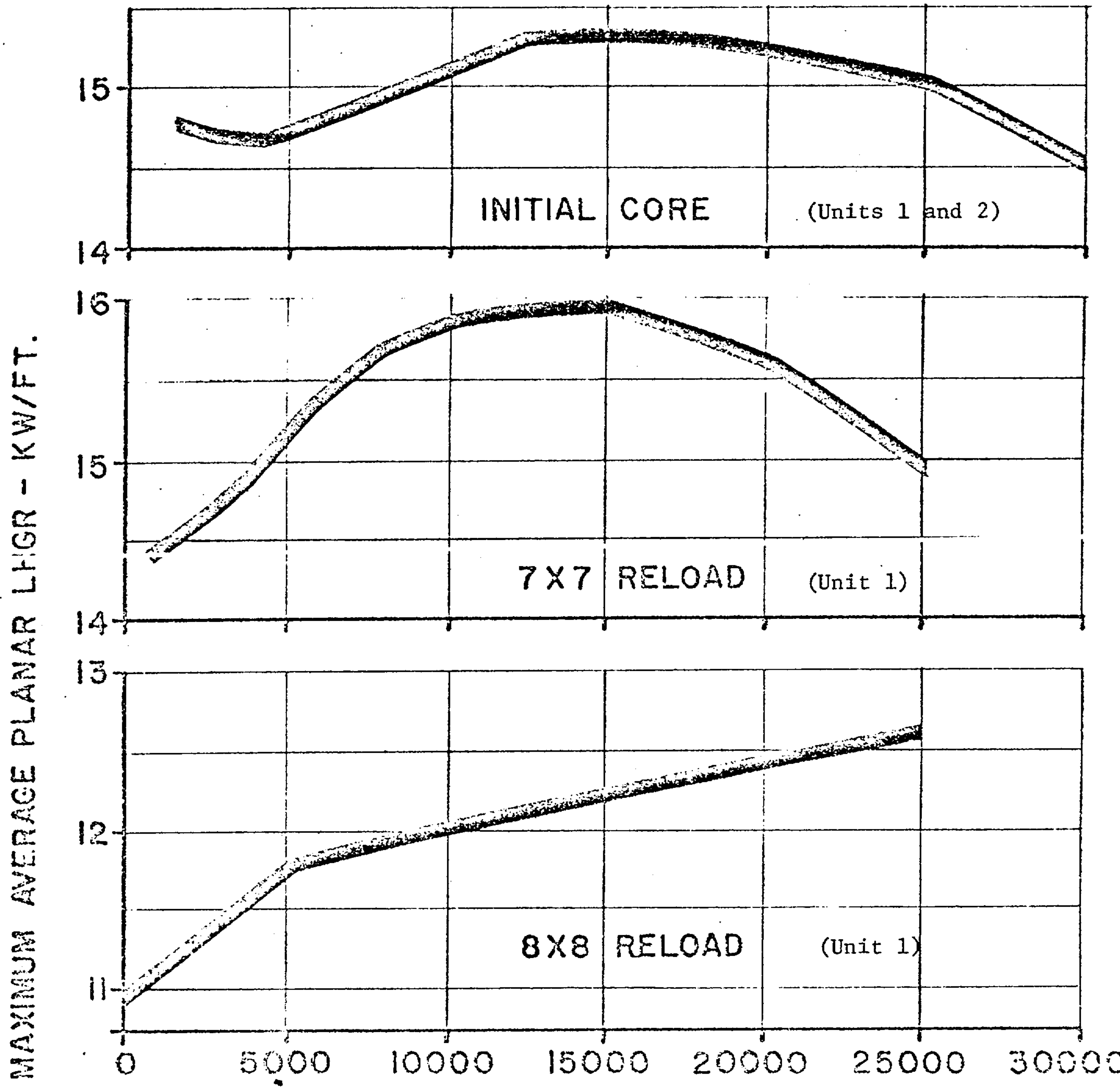

FIGURE 3.5.1 MAXIMUM ALLOWABLE PLANAR LHGR APPLICABLE TO QUAD-CITIES INITIAL AND RELOAD FUEL

19

3.5 LIMITING CONDITIONS FOR OPERATION

J. Local LHGR

During steady state power operation, the linear heat generation rate (LHGR) of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following equation.

$$LHGR_{max} \leqslant LHGR_d \left[1 - \left(\frac{\Delta P}{P}\right)_{max} \left(\frac{L}{L_T}\right) \right]$$

$LHGR_d$ = Design LHGR

= 17.5 kw/ft, 7X7 fuel assemblies

19 = 13.4 kw/ft, 8X8 fuel assemblies

$\left(\frac{\Delta P}{P}\right)_{max}$ = Maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7X7 fuel

= .024 reload 1, 8X8 fuel

= .028 reload 1, mixed oxide fuel

L_T = Total Core Length

= 12 ft

L = Axial distance from bottom of core

4.5 SURVEILLANCE REQUIREMENTS

19 J. Local LHGR

Daily during reactor power operation, the local LHGR shall be checked.

I. Average Planar LHGR

19 This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2300°F limit specified in the Interim Acceptance Criteria (IAC) issued in June 1971 considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the IAC limit.

19 The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference 1 as modified by Reference 2, and authorized in Reference 3.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any rod is less than the design linear heat generation rate even if fuel pellet densification is postulated. The power spike penalty specified is based on that presented in Reference 4, and assures a linearly increasing variation in axial gaps between core bottom and top, and assures with a 95% confidence, that no more than one fuel rod exceeds the design linear heat generation rate due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining $\Delta P/P$ in accordance

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19
(1) NEDM-10735, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Aug. 1973.
(2) NEDO-20181, "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," Nov. 1973.
(3) D.J. Skovholt (USAEC) Letter to J.S. Abel (CE Co.) Dec. 5, 1973.
(4) NEDM-10735, "Fuel Densification Effects on General Electric Boiling Water Reactor Fuel," Section 3.2.1, Supplement 6, Aug. 1973.

3.5 Limiting Condition for Operation Bases (Cont'd)

19 with References 5 and 6.

4.5 Surveillance Requirements Bases:

The testing interval for the core and containment cooling systems is based on a quantitative reliability analysis, judgment, and practicality. The core cooling systems have not been designed to be fully testable during operation. For example, the core spray final admission valves do not open until reactor pressure has fallen to 350 psig. Thus, during operation, even if high drywell pressure were simulated, the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems, the components which make up the system, i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure
3 that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The surveillance requirements to ensure the discharge piping of the core spray, LPCI mode of the RHR, HPCI, and RCIC systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor

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(5) J.A. Hinds (GE) Letter to V.A. Moore (USAEC), "Plant Evaluation with GE
19 GEGAP-III," Dec. 12, 1973.

(6) USAEC Report, "Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels," Dec. 14, 1973.

5.0 DESIGN FEATURES

5.1 Site

The Quad-Cities Nuclear Power Station which consists of a tract of land of approximately 404 acres is located about 3 miles north of Cordova, Illinois, Rock Island County, Illinois. The tract is situated in portions of Section 7, 8, 17 and 18 of Township 20 North, Range 2 East.

5.2 Reactor

19 A. The core shall consist of not more than 724 fuel assemblies.

B. The reactor core shall contain 177 cruciform-shaped control rods. The control material shall be boron carbide power (B_4C) compacted to approximately 70% of theoretical density.

5.3 Reactor Vessel

The reactor vessel shall be as described in Table 4.1.1 of the SAR. The applicable design codes shall be as described in Table 4.1.1 of the SAR.

5.4 Containment

A. The principal design parameters and applicable design codes for the primary containment shall be as given in Table 5.2.1 of the SAR.

B. The secondary containment shall be as described in Section 5.3.2 of the SAR and the applicable codes shall be as described in Section 12.1.1.3 of the SAR.

C. Penetrations to the primary containment and piping passing through such penetrations shall be designed in accordance with standards set forth in Section 5.2.2 of the SAR.

5.5 Fuel Storage

A. The new fuel storage facility shall be such that the K_{eff} dry is less than 0.90 and flooded is less than 0.95.

B. The K_{eff} of the spent fuel storage pool shall be less than or equal to 0.90.

5.6 Seismic Design

The reactor building and all contained engineered safeguards are designed for the maximum credible earthquake ground motion with an acceleration of 24 percent of gravity. Dynamic analysis was used to determine the earthquake acceleration, application to the various elevations in the reactor building.

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

SAFETY EVALUATION BY THE DIRECTORATE OF LICENSING
SUPPORTING AMENDMENT NO. 10 TO LICENSE NO. DPR-29
(CHANGE NO. 19 TO APPENDIX A TECHNICAL SPECIFICATIONS)
COMMONWEALTH EDISON COMPANY
QUAD-CITIES NUCLEAR POWER STATION UNIT 1
DOCKET NO. 50-254

INTRODUCTION

By application dated February 28, 1974, Commonwealth Edison Company (CE) requested authorization to operate Quad-Cities Unit 1 with Reload 1 fuel assemblies in the core. According to CE's plan, approximately 88 uranium and 5 mixed oxide (PuO_2 - UO_2) Reload 1 assemblies will replace an equal number of assemblies presently in the core. Reload 1 will consist of 7 x 7 fuel assemblies with a 2.30 w/o average enrichment, and 8 x 8 fuel with a 2.50 w/o average enrichment or a combination of 8 x 8 and 7 x 7 assemblies. The request to use mixed oxide fuel is being considered as a separate matter.

Supplements to the application were submitted by letters dated June 15, 1973, December 14, 1973, January 28, 1974, and March 18, 1974.

The safety analyses of the reload submittal by the licensee include evaluation of the effect of the reload on previously analyzed conditions during normal operations, operational transients and postulated accidents. Included in these analyses are considerations of the applicability of existing technical specifications and the need for revisions. The evaluation included consideration of the reload fuel bundles of the presently used 7 x 7 array, the new design reload assemblies in an 8 x 8 array, a 7 x 7 segmented test rod bundle, and the characteristics of the core 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 was evaluated by the Regulatory staff in a previous report[(1)]. The use of 8 x 8 fuel for reloads also was reviewed by the Advisory Committee on Reactor Safeguards and discussed in its report dated February 12, 1974.

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

The 7 x 7 Reload 1 fuel design, except the segmented rod, is the same as the fuel presently in the core. The applicant's methods of analysis, used and approved for the initial core, are therefore applicable to the 7 x 7 Reload 1 fuel assemblies.

The 7 x 7 segmented test rod (STR)[2] fuel assembly will be in a low power region of the core and, therefore, the licensee does not expect the assembly to approach any performance limit. The impact of this bundle on neighboring bundles will be negligible since its nuclear characteristics have been maintained essentially the same as the 7 x 7 Reload 1 bundles. The fuel in the STR bundle will contain the same enrichment as the rods which they replace in the standard reload bundle. The reactivity of the segmented test rod bundle has been reduced slightly due to a smaller inventory of UO_2. Based on these considerations we concluded that the methods of analysis used and approved for the initial core are applicable to the 7 x 7 STR fuel assembly. Therefore the following evaluation for 7 x 7 reload fuel applies equally to the 7 x 7 STR fuel bundle.

EVALUATION

The reference core consists of 724 initial 7 x 7 fuel assemblies, which are to be replaced with up to sixty 7 x 7 Reload 1 assemblies, which are identical to the initial fuel assemblies and up to eighty-eight 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 patterns 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 Quad-Cities Unit 1.

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 Quad-Cities Unit 1 8 x 8 fuel assemblies are of similar design and material to the 7 x 7 fuel assemblies which have successfully been operated at Quad-Cities Unit 1. 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.

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(2) NEDE-20236 "STR Bundle Submittal Quad-Cities 1 Segmented Test Rod Bundle" (Proprietary) dated January 1974.

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 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 8 x 8 reload fuel for Quad-Cities Unit 1 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 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 the 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 8 x 8 reload fuel for Quad-Cities Unit 1 is acceptable and will provide an increased margin of safety as compared with the previously operated 7 x 7 assemblies.

A. Proposed Changes to Technical Specifications

Since the performance characteristics of the Reload 1 bundles are similar to the previously authorized loading, the safety limits and limiting safety system settings presently specified in the Technical Specifications are applicable. With one exception, the limiting conditions of operation also are unchanged. This exception is the average planar and local linear heat generation rate (LHGR) limits.

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) and new 7 x 7 fuel 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 utilize the 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 Quad-Cities Unit 1 and approved by Change 16 to DPR-29 dated May 14, 1974. That evaluation concluded that the use of the new model has essentially 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. Standby Liquid Control System Reactivity

Original analysis of the standby liquid control system indicated that a solution of 600 ppm of boron would provide a K_{eff} of less than 0.95 for the Quad Cities Unit 1 core. Reanalysis of the effectiveness of the standby liquid control system in the reloaded Unit 1 core indicates that 600 ppm of boron will provide a K_{eff} of 0.97 in the cold, xenon-free condition of the Unit 1 core. Even though the reanalysis results in a higher K_{eff} (0.97 vs 0.95) for the poisoned core, the original design basis for the standby liquid control system is still met in that the system is designed to bring the reactor core to a shutdown condition at any time during core life, independent of the control rod system.

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

C. Abnormal Operational Transients

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 fuels 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 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. Amendment No. 8 dated May 24, 1974 for the license (DPR-29) for Quad Cities 1 authorized the replacement of one electromatic relief valve with a Target Rock safety relief valve. The effect of this replacement on the allowable operation power level is discussed in the above amendment.

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 Quad Cities Unit 1. 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 Quad Cities Unit 1 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 are 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 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 released 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 there is reasonable assurance that the health and safety of the public will not be endangered by the proposed refueling and subsequent operation with Reload 1 and with the proposed modifications to the Technical Specifications.

/s/

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

Original Signed by:
Dennis L. Ziemann

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

Date: JUN 05 1974

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-254

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF CHANGES TO TECHNICAL SPECIFICATIONS OF FACILITY OPERATING LICENSE

No request for a hearing or petition for leave to intervene having been filed following publication of the notice of proposed action in the Federal Register on March 22, 1974 (39 F.R. 10928), the Atomic Energy Commission (the Commission) has issued Change No. 19 to the Technical Specifications of Facility Operating License No. DPR-29 to the Commonwealth Edison Company (the licensee, acting for itself and on behalf of Iowa-Illinois Gas & Electric Company). This change, effective immediately, authorizes the licensee to operate the Quad-Cities Nuclear Power Station Unit 1 using a partial loading of 8 x 8 fuel assemblies (containing U-235), and one 7 x 7 fuel assembly containing segmented test rods, and would also authorize changes to the limiting conditions for operation associated with fuel densification for the new 8 x 8 and 7 x 7 fuels. The licensee is presently licensed to possess and operate the Quad-Cities Nuclear Power Station Unit 1, located in Rock Island County, Illinois, at power levels up to 2511 MWt using a full core of 7 x 7 fuel assemblies (containing U-235).

The Commission has found that the application for the above action filed by Commonwealth Edison dated February 28, 1974, including the supplements dated January 28, 1974 and February 7, 1974 (filed prior to the basic application) and subsequent supplement dated March 18, 1974, complies

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with the requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's regulations published in 10 CFR Chapter I.

The Commission's Directorate of Licensing has completed its evaluation of the above action and a Safety Evaluation is being issued concurrently with this notice concluding that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of the facility with the 8 x 8 fuel and the related changes to the Technical Specifications as authorized by Change No. 19, which is incorporated in License No. DPR-29 as Amendment No. 10 thereto.

Copies of (1) Amendment No. 10 with Change No. 19 to the Technical Specifications of Facility Operating License No. DPR-29, (2) the Commission's concurrently issued Safety Evaluation, (3) "Technical Report on the General Electric Company 8 x 8 Fuel Assembly" dated February 5, 1974, by the Directorate of Licensing, (4) Report NEDO-20103 - "General Design Information for General Electric Boiling Water Reactor Reload Fuel Commencing in Spring 1974", and (5) the Advisory Committee on Reactor Safeguard's letter of February 12, 1974, "Report on General Electric 8 x 8 Fuel Design for Reload Use" are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C., and at the Moline Public Library at 504 - 17th Street, Moline, Illinois 61265. Single copies of items 1, 2, 3 and 5 may be obtained upon request sent to the Deputy Director for Reactor Projects, Directorate of Licensing, U. S. Atomic Energy Commission, Washington, D. C. 20545.

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Dated at Bethesda, Maryland, JUN 0 5 1974

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by:
Dennis L. Ziemann

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

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