Docket No. 50-254

Correct Date - 5-3-77 GD.

3/3/77

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Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your request dated January 21, 1977, and a supplement thereto dated April 25, 1977, the Commission has issued the enclosed Amendment No. 41 to Facility Operating License Noo DPR-29 for Unit No. 1 of the Quad Cities Nuclear Power Station.

This amendment (1) authorizes operation with additional 8 x 8 fuel assemblies, (2) incorporates revised MCPR limits in response to the plant specific analysis for reload 3 and (3) modifies License Condition 3.C to reflect End-of-Cycle scram reactivity conditions for reload 3.

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

Sincerely,

Original signed by
Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 41 to License No. DPR-29
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

Const. 1
GD

*NOTE: FOR PREVIOUS CONCURRENCES, SEE ATTACHED YELLOW.

OFFICE	DOR:ORB #2 PWO	DOR:ORB #2	OELD	DOR:RSB/OT	DOR:ORB #2	
SURNAME	PO'Connor:ro	RMDiggs		RBaer RB	DKDavis	
DATE	5/3/*77	5/4/77	/ /*	5/3/77	5/3/77	

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Ross
TBAbernathy
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OFFICE	DOR:ORB #2	DOR:ORB #2	OELD	DOR:RSB/OT	DOR:ORB #2	
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DATE	4/29/77	/ /77	5/2/77	/ /77	/ /77	

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

May 3, 1977

Docket No. 50-254

Commonwealth Edison Company
ATTN: Mr. R. L. Bolger
Assistant Vice President
Post Office Box 767
Chicago, Illinois 60690

Gentlemen:

In response to your request dated January 21, 1977, and a supplement thereto dated April 25, 1977, the Commission has issued the enclosed Amendment No. 41 to Facility Operating License No. DPR-29 for Unit No. 1 of the Quad Cities Nuclear Power Station.

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Sincerely,

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Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 41 to
   License No. DPR-29
2. Safety Evaluation
3. Notice

cc w/enclosures:
See next page

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Rock Island County Court House
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Agency
Federal Activities Branch
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cc w/enclosures and copy of
CECo filings dtd. 1/21/77
& 4/25/77:
Department of Public Health
ATTN: Chief, Division of
Radiological Health
535 West Jefferson
Springfield, Illinois 62706

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-254

QUAD CITIES UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 41
License No. DPR-29

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated January 21, 1977, as supplemented on April 25, 1977, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes 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-29 are hereby amended to read as follows:

   3.B <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 41, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

   3.C <u>Restrictions</u>

   Reactor power level shall be limited to maintain pressure margin to the safety valve set points during the worst case pressurization transient. The magnitude of the power limitation, if any, and the point in the cycle at which it shall be applied is specified in the Reload No. 3 licensing submittal for Quad Cities Unit 1 (NEDO-21489). Plant operation shall be limited to the operating plan described therein.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 3, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 41

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

Replace the following pages of the Technical Specifications contained in Appendix A of the above-indicated license with the attached pages bearing the same numbers. The changed areas on the revised pages are reflected by a marginal line.

Remove Pages	Insert Pages
1.1/2.1-2	1.1/2.1-2
1.2/2.2-2	1.2/2.2-2
1.2/2.2-3	1.2/2.2-3
3.2/4.2-8	3.2/4.2-8
3.5/4.5-10	3.5/4.5-10
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15

D. **Reactor Water Level (Shutdown Condition)**

Whenever the reactor is in the shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than that corresponding to 12 inches above the top of the active fuel when it is seated in the core.

curve in Figure 2.1-2, at which point the actual peaking factor value shall be used.

LTPF = 3.06 (7 x 7 fuel assemblies)
3.03 (8 x 8 fuel assemblies)

2. APRM Flux Scram Trip Setting (Refueling or Startup and Hot Standby Mode)

   When the reactor mode switch is in the Refuel or Startup Hot Standby position, the APRM scram shall be set at less than or equal to 15% of rated neutron flux.

3. IRM Flux Scram Trip Setting

   The IRM flux scram setting shall be set at less than or equal to 120/125 of full scale.

4. When the reactor mode switch is in the startup or run position, the reactor shall not be operated in the natural circulation flow mode.

B. **APRM Rod Block Setting**

The APRM rod block setting shall be as shown in Figure 2.1-1 and shall be:

$$S \leq (.65W + 43)(LTPF/TPF)$$

The definitions used above for the APRM scram trip apply.

C. Reactor low water level scram setting shall be $\geq$ 143 inches above the top of the active fuel at normal operating conditions.

D. Reactor low water level ECCS initiation shall be 83 inches (+4 inches/-0 inch) above the top of the active fuel at normal operating conditions.

E. Turbine stop valve scram shall be $\leq$ 10% valve closure from full open.

F. Turbine control valve fast closure scram shall initiate upon actuation of the fast closure solenoid valves which trip the turbine control valves.

G. Main steamline isolation valve closure scram shall be $\leq$ 10% valve closure from full open.

H. Main steamline low-pressure initiation of main steamline isolation valve closure shall be $\geq$ 850 psig.

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 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 limits the pressure to approximately 1100 psig (References 1, 2, and 3). 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 (References 4, 5, 6, and 7) 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 relief valves during the postulated full closure of all MSIVs without direct (valve position switch) scram. Credit is taken for the neutron flux scram, however. The pressure at the bottom of the vessel peaks at less than 1325 psig. The indirect flux scram and safety valve actuation, therefore, provide adequate margin below the peak allowable vessel pressure of 1375 psig.

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

References

1. SAR, Section 11.2.2.
2. 'Quad-Cities 1 Nuclear Power Station First Reload License Submittal,' Section 6.2.4.2, February 1974.
3. GE Topical Report NEDO-20693, 'General Electric Boiling Water Reactor Reload No. 1 Licensing Submittal for Quad-Cities Nuclear Power Station Unit 2,' December 1974.
4. SAR Section 4.4.3, 'Quad-Cities 1 Nuclear Power Station First Reload License Submittal,' Section 6.2.4.2, February 1974.
5. Dresden 3 Special Report No. 29, 'Transient Analysis for Cycle 2'.
6. Letter to D. J. Skovholt from J. S. Abel, October 18, 1973, Subject: Scram Reactivity Limitations for Dresden Units 2 and 3 and Quad-Cities Units 1 and 2.
7. Dresden Station Special Report No. 29, Supplement B.

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 neutron flux 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 available as backup protection to the high flux scram which was analyzed (Reference 1) in Section 4.4.3 of the SAR, reexamined in Reference 2 and the reload license submittal for each subsequent cycle. If the high flux scram were to fail, a high-pressure scram would occur at 1060 psig.

References

1. 'Quad-Cities/Nuclear Power Station First Reload License Submittal,' Section 6.2.4.2, February 1974.

2. Dresden Station Special Report No. 29, Supplement B.

so that none of the activity released during the refueling accident leaves the reactor building via the normal ventilation stack but that all the activity is processed by the standby gas treatment system.

The instrumentation which is provided to monitor the postaccident condition is listed in Table 3.2-4. The instrumentation listed and the limiting conditions for operation on these systems ensure adequate monitoring of the containment following a loss-of-coolant accident. Information from this instrumentation will provide the operator with a detailed knowledge of the conditions resulting from the accident; based on this information he can make logical decisions regarding postaccident recovery.

The specifications allow for postaccident instrumentation to be out of service for a period of 7 days. This period is based on the fact that several diverse instruments are available for guiding the operator should an accident occur, on the low probability of an instrument being out of service and an accident occurring in the 7-day period, and on engineering judgment.

The normal supply of air for the control room ventilation system comes from outside the service building. In the event of an accident, this source of air may be required to be shut down to prevent high doses of radiation in the control room. Rather than provide this isolation function on a radiation monitor installed in the intake air duct, signals which indicate an accident, i.e., high drywell pressure, low water level, main steamline high flow, or high radiation in the reactor building ventilation duct, will cause isolation of the intake air to the control room. The above trip signals result in immediate isolation of the control room ventilation system and thus minimize any radiation dose.

References

1. GE Topical Report NEDO-20693, 'General Electric Boiling Water Reactor Reload No. 3 Licensing Submittal for Quad-Cities Nuclear Power Station (Unit 1)', Section 6.3.3.2, November, 1976.

2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits.

$$LHGR_{max} \quad LHGR_d \; I \quad (\Delta P/P)_{max}(L/L_t)$$

$LHGR_d$ = design LHGR

where:

= 17.5 kW/ft, 7 x 7 fuel assemblies

= 13.4 kW/ft, 8 x 8 fuel assemblies

$(\Delta P/P)_{max}$ = maximum power spiking penalty

= .035 initial core fuel

= .029 reload 1, 7 x 7 fuel

= .022 reload, 8 x 8 fuel

= .028 reload 1, mixed oxide fuel

L_t = total core length

= 12 feet

L = Axial distance from bottom of core

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation MCPR shall be greater than or equal to

1.35 (7 x 7 fuel)
1.32 (8 x 8 fuel)

at rated power and flow. If at any time during operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. If the steady-state MCPR is not returned to within the prescribed limits within 2 hours, the reactor shall be brought to the cold shutdown condition within 36 hours. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. For core flows other than rated, these nominal values of MCPR shall be increased by a factor of K_f, where K_f is as shown in Figure 3.5-2.

K. Minimum Critical Power Ratio (MCPR)

The MCPR shall be determined daily during steady-state power operation above 25% of rated thermal power.

H. Condensate Pump Room Flood Protection

See Specification 3.5.H.

I. Average Planar LHGR

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 secondarily dependent 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 cladding 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. The maximum average planar LHGR's shown in Figure 3.5-1 are based on calculations employing the models described in Reference 2 as modified by Reference 3 and authorized in Reference 4.

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 5 and assumes 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 Δ/P in accordance with References 6 and 7.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this specification were selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state as well as uncertainties in the critical power correlation itself. These values also assure that operation will be such that the initial condition assumed for the LOCA analysis, an MCPR of 1.18, is satisfied. For any of the special set of transients or disturbances caused by single operator error or single equipment malfunction, it is required that design analyses initialized at this steady-state operating limit yield a MCPR of not less than that specified in Specification 1.1.A at any time during the transient, assuming instrument trip settings given in Specification 2.1. For analysis of the thermal consequences of these transients, the limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. The results apply with increased conservatism while operating with MCPR's greater than specified.

The most limiting transients with respect to MCPR are generally:

a) Rod withdrawal error

b) Turbine trip without bypass

c) Loss of feedwater heater

Several factors influence which of these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycles reload licensing submittal specifies the limiting transients for a given exposure increment for each fuel type. The values specified as the Limiting Condition of Operation are conservatively chosen as the most restrictive over the entire cycle for each fuel type.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This assures that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

References

1. I. M. Jacobs and P. W. Marritt, GE Topical Report APED-5736, 'Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards,' April 1969.

A. GE Topical Report NEDM-10735, 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' August 1973.

3. GE Topical Report NEDO-20181, 'GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods,' November 1973.

4. D. J. Skovholt, USAEC, Letter to J. S. Abel, CECo, December 5, 1973.

5. GE Topical Report NEDM-10735, 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' Section 3.2.1, Supplement 6, August 1973.

6. J. A. Hinds, GE, Letter to V. A. Moore, USAEC, 'Plant Evaluation with GE GEGAP-III,' December 12, 1973.

7. USAEC Report, 'Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels,' December 14, 1973.

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION
SUPPORTING AMENDMENT NO.41 TO FACILITY LICENSE NO. DPR-29
COMMONWEALTH EDISON COMPANY
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY
QUAD CITIES NUCLEAR POWER STATION UNIT 1
DOCKET NO. 50-254

1.0 INTRODUCTION

By letter dated January 21, 1977, and a supplement thereto dated April 25, 1977, Commonwealth Edison (the licensee) requested an amendment to Facility Operating License No. DPR-29. The amendment would modify the license and Technical Specifications for Quad Cities Station Unit No. 1 to permit operation.

(1) with additional 8x8 fuel assemblies (Reload 3), as requested in their application dated January 21, 1977;

(2) incorporating revised MCPR limits in response to the plant specific analysis for Reload 3; and

(3) with License Condition 3.C modified to reflect end-of-cycle scram reactivity conditions for Reload 3.

During our review of the proposed technical specifications we determined that certain changes were necessary to conform with Regulatory requirements. These changes have been accepted by the licensee.

Discussion

The reference core loading for Reload 3 is indicated in Table 1. This loading differs from the previously authorized Reload 2 by the replacement of 4 Reload-2 8x8 assemblies and 180 initial 7x7 assemblies by 184 Reload-3 8x8 fuel assemblies. The reload assemblies are scatter loaded throughout the core. The acceptability of the neutronic, thermal-hydraulic, and mechanical design of 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 Quad Cities 1 was evaluated and approved by Amendment No. 10 to Facility Operating License No. DPR-29 dated June 5, 1974.

Our safety evaluation of this reload (Reload No. 3) for the Quad Cities Unit 1 core is based on the licensee's application[3] as supplemented and on information contained in a GE topical report, NEDO-20360[4] referenced in the application.

Table 1

FUEL TYPE AND NUMBER

Fuel Type		Number
Initial		320
Reload-1	(7D230)	22
	(STR)	1
	(8D250)	36
	(Pu)	5
Reload-2	(8D250)	104
	(8D262)	52
Reload-3	(8D250)	184
Total		724

2.0 EVALUATION

2.1 Nuclear Characteristics

The reload information presented in the licensing submittal closely follows the guidelines of Appendix A of the Generic Reload Licensing Application, NEDO-20360[4]. The NRC Staff has reviewed the contents of the latest supplements to this generic topical report and has found them acceptable for use in connection with the information submitted by the licensee in support of his application for this core reload. The information contained in NEDO-20360 through Supplement No. 4 does not alter the conclusions and approvals as stated in Reference 5.

Up to 184 8x8 reload fuel bundles with an average U-235 enrichment of 2.50 wt/% will be loaded throughout the core. Each of the reload bundles contains several gadolinia bearing fuel rods. The gadolinia rods have been incorporated in order to reduce the amount of excess reactivity early in life and also to aid in controlling local power peaking. The core contains a total of 724 fuel assemblies. Thus, approximately 25 percent of the fuel bundles are being replaced for this reload. Previously, for Reload 2, a total of 160 fresh 8x8 reload assemblies were loaded into the core. Of these 108 had an average U-235 enrichment of 2.50/wt% while 52 had a U-235 enrichment of 2.62 wt%. All of the Reload 2 fuel assemblies contained several fuel rods bearing gadolina.

The Cycle 4 core loading pattern may be described as follows:

(1) the two rows and columns of fuel bundle locations intersecting the center of the core will not be loaded with any Reload 3 fuel, (2) in the core interior, alternating unit cell locations will contain one or two fresh fuel assemblies in a checkerboard manner, (3) all of the bundle on the core periphery will be from the initial core, (4) no more than one fuel bundle from each of the previous reloads surround any given interior control rod (except for the two central rows and columns of fuel bundles), and (5) the core loading results in quadrant symmetry. The 8x8 reload fuel for the Reload 3 core is, therefore, basically scatter loaded.

The information in Reference 3 indicates that the nuclear characteristics of the Cycle 4 core, consisting of both the reload 8x8 fuel and the previously exposed 7x7 and 8x8 fuel, are very similar to previous cores. Typical nuclear characteristics of the reloaded core are given in Table 5-1 of Reference 3. The void coefficient of reactivity with a core average void content of 36.3 percent varies from -10.55×10^{-4} to -10.92×10^{-4} $\Delta K/K/\%V$. The Doppler coefficient, at a fuel average temperature of 650°C, varies from -1.123×10^{-5} to -1.21×10^{-5} $\Delta K/K/^{\circ}F$. Thus, based on our review of the information presented in the Quad Cities Unit No. 1 licensing submittal and the generic 8x8 reload topical report, it is concluded that the fuel temperature and void dependant behavior of the reconstituted core will not differ significantly from cores which have been previously reviewed and approved by the Commission in connection with past operating cycles of the Quad Cities Unit No. 1 reactor.

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 single most reactive control rod fully withdrawn and with all other rods fully inserted. For Cycle 4 the minimum shutdown margin at any time during the cycle is 1.22%Δk. The shutdown margin at the beginning of Cycle 4 is calculated to be 1.55%Δk with the most reactive control rod fully withdrawn.

The maximum shutdown margin loss (.0033Δk) during the cycle accounts for the potential effect of B_4C settling in all inverted poison tubes present in the core. The information presented in Reference 3 indicates that a boron concentration of 600 ppm in the moderator will make the reactor subcritical by at least 0.03Δk at 20^{o}C, xenon free. Therefore, the alternate shutdown requirement of the General Criteria is met by the Standby Liquid Control System. We find these results acceptable.

The Technical Specification requirement for the storage of fuel for Quad Cities Unit No. 1 is that the effective multiplication factor, k_{eff}, of the fuel as stored in the fuel storage rack is equal to or less than 0.90 for normal storage conditions. This requirement is met if the uncontrolled infinite multiplication factor, k_∞ of a fuel bundle in the reactor core configuration is less than or equal to 1.30.[4] The 8x8 8D250 fuel bundle, at both the zero exposure and the peak reactivity point, has an uncontrolled k_∞ of 1.236. Therefore, the Technical Specifications requirement for fuel storage subcriticality is satisfied.

The Cycle 4 exposure dependent scram reactivity curves used for the analysis of abnormal operating transients are shown in Figure 6-6 of Reference 3. The two curves shown correspond to 1000 Mwd/T before EOC4 and EOC4. These scram curves include acceptable design conservatism factors of 1.25 for the void coefficient and 0.80 for the scram reactivity function.

Thus, based on our review of the information presented in the Quad Cities Unit No. 1 licensing submittal and the generic 8x8 reload topical report we conclude that the nuclear characteristics (e.g., scram reactivity, void coefficient of reactivity and Doppler coefficient of reactivity) and performance of the reconstituted core for the Reload 3 cycle will not differ significantly from previously analyzed and approved Quad Cities Unit 1 fuel cycles and are acceptable.

2.2 Mechanical Design

The reload fuel consists of up to 184 GE 8x8 fuel bundles with average U-235 enrichment of 2.50 wt/%. These fuel assemblies have the same mechanical design and enrichments as the 8D250 fuel assembly described in the 8x8 generic reload report[4]. This generic report has been reviewed and with some modifications, has been found acceptable for use in connection with BWR-3 reactors containing 8x8 reload fuel, when supplemented with information required by the Staff status report[5] on the GE generic report evaluation.

In addition, all of the Reload 3 fuel incorporates finger springs for controlling bypass flow at the interface of the channel and fuel bundle lower tie plate. This device has previously been used satisfactorily in connection with General Electric's initial and reload fuel for all BWR-4 cores as well as for several BWR-3 plants. Finger springs have been used previously at Quad Cities Unit No. 1, during Cycle 3, on 12 8x8 reload assemblies. Inspection of more than 900 fuel assemblies employing finger springs in operating BWR has demonstrated acceptable operating performance of this fuel mechanical design feature.

On the basis of our review of the generic 8x8 reload report, the reload submittal and current operating experience with the 8x8 reload design in similar plants, it is concluded that the Reload 3 fuel for Quad Cities Unit 1 has an acceptable mechanical design.

2.3 Thermal-Hydraulics

The generic 8x8 reload topical report[4] and the General Electric Thermal Analysis Basis (GETAB)[6] are referenced to provide the description of the thermal-hydraulic methods which were used to calculate the thermal margins. Application of GETAB, based on the Minimum Critical Power Ratio (MCPR) concept, was used to establish the:

(1) fuel cladding integrity safety limit,

(2) limiting conditions of operating such that the safety limit is not exceeded for normal operation and abnormal operational transients, and

(3) limiting conditions of operation such that the initial conditions assumed in the accident analyses are satisfied.

The Staff has reviewed [7] the GETAB report and has found it acceptable for use in the above applications for 8x8 and 7x7 fuel assemblies.

The Quad Cities Unit No. 1, Cycle 4 thermal limits based on the GETAB report and the plant specific information provided by the licensee have been reviewed. The Staff evaluation of these limits is reported herein.

2.3.1 Fuel Cladding Integrity Safety Limit Minimum Critical Power Ratio (MCPR)

A critical power ratio (CPR) is defined as the ratio of that assembly power which causes some point in the assembly to experience transition boiling to the actual assembly power. The minimum critical power ratio (MCPR) is the critical power ratio corresponding to the most limiting fuel assembly in the core. The fuel cladding integrity safety limit MCPR is 1.06 for both 7x7 and 8x8 fuel types. This safety limit, based on the GETAB statistical analysis [(4)], assures that 99.9% of the fuel rods in the core are not expected to experience transition boiling for abnormal operational transients. The uncertainties in the core operating parameters, plant system operating parameters and the GEXL correlation, when combined with the design relative bundle power histogram for the core, form the basis of the GETAB statistical determination of the safety limit MCPR. The tabulated list of uncertainties for Quad Cities Unit No. 1 during Cycle 4 are (the same as Cycle 3)at least as conservative as those used in the GETAB report (revision to Table IV-1 of NEDO-10958[(8)]). For example,the Cycle 4 analysis includes an increase in the "TIP Reading" standard deviation from 6.3 to 8.7 percent. The increase in uncertainty for the subject reload is a consequence of the increase in uncertainty in the measurement of local power in the reloaded core.

The generic core selected for the GETAB statistical analysis is a typical 251/764 core while Quad Cities Unit 1 is a 251/724 core. The generic GETAB statistical analysis results are conservative, however, since the core bundle power histogram used in the generic GETAB application has more high power bundles than the most adverse bundle power distribution expected at any time during the fourth cycle of operation of Quad-Cities Unit No. 1. This results in a conservative value of the safety limit MCPR which meets the 99.9% criterion.

We conclude that the proposed fuel integrity safety limit MCPR of 1.06 is acceptable for both the 7x7 and 8x8 fuel in the Quad Cities Unit No. 1 reactor core during Cycle 4 (Reload-3).

2.3.2 Operating Limit MCPR

Various transient events will reduce the MCPR below the steady-state operating value. To assure that the fuel cladding safety limit MCPR of 1.06 is not violated during anticipated abnormal operational transients, the most limiting transients have been re-analyzed for Cycle 4 to determine which results in the largest reduction in the critical power ratio (i.e. ΔCPR). The licensee has submitted[(3)] the results of analyses of those transients which produce the most significant decrease in MCPR. The types of anticipated abnormal operational transients evaluated were reactor pressure increase, feedwater temperature decrease, coolant flow increase, etc.

A GE study[6] has shown that the required operating MCPR varies with the axial and local (pinwise) power peaking distribution. Axial peaking in the middle or upper portion of the core results in higher required MCPR's than peaking in the lower portion of the core. In the analyses the axial power peaking was assumed to be representative of beginning-of-cycle conditions, located at the core midplane, with an axial peak-to-average ratio of 1.40.

The bundle R-factors, which are a function of the local power peaking distribution, assumed in the GETAB analysis are also representative of a beginning of cycle condition. The R-factor values used were 1.100 for 7x7 fuel and 1.094 for the 8x8 fuel. During the cycle the local peaking, and therefore the R-factor, is reduced while the peak in the axial shape moves toward the bottom of the core. The amount by which the R-factor decreases from beginning to end-of-cycle would, by itself, increase the required operating limit MCPR by approximately 1 percent. This adverse effect on the MCPR is offset, however, by a beneficial relocation of the axial peak to below the core midplane. Overall conservatism was applied in the determination of the required operating limit MCPR, since the assumed axial and local peaking were representative of the beginning of cycle, which provides the most adverse consistent set of axial and local peaking conditions.

The most limiting abnormal operational transient occurring at any time during Cycle 4 from rated conditions in the categories discussed above is 1) a rod withdrawal error for the 7x7 fuel, which results in a maximum ΔCPR of 0.24 and 2) a turbine trip with failure of the bypass valves for the 8x8 fuel, which results in a maximum ΔCPR of 0.21.

Addition of these ΔCPR's to the safety limit MCPR would normally provide the minimum operating limit MCPR for each fuel type, required to avoid violation of the safety limit, should these limiting transients occur. The licensee has therefore proposed MCPR operating limits of 1.30 and 1.27 for the 7x7 and 8x8 fuel types respectively for Cycle 4. However, the licensee reports in the reload submittal[3] that the most severe fuel loading error, consisting of a fresh 8x8 bundle loaded in a core position analyzed for a high burnup 8x8 assembly, results in a ΔCPR of 0.26 which exceeds the ΔCPR associated with the most limiting abnormal operational transient for both 7x7 and 8x8 fuel types. This fuel loading error could, therefore, decrease the MCPR below the safety limit MCPR (i.e. to 1.01) if the operating limit were based solely on the consideration of anticipated operational transients.

The Staff has the fuel loading error under generic review. Until this issue is resolved, the Staff in the interim, requires that the operating limit MCPR; proposed by the licensee be increased an additional .05 for all fuel types to account for the possibility of a fuel loading error.

Thus, based on the analyses of both the most severe abnormal operational transients and the fuel loading error, we require that the operating limit MCPR be 1.35 for 7x7 fuel and 1.32 for 8x8 fuel to avoid violating the safety limit in the event of a fuel loading error from rated conditions. The licensee has agreed to increase the Cycle 4 operating limit MCPR to these values.

The abnormal operational transients were evaluated with scram reactivity insertion rates that included a design conservatism factor of .80. The analyses also included a design conservatism factor of 1.25 for the void coefficients. These design conservatism factors are acceptable as are the initial conditions used for the most severe abnormal operational transients. The initial MCPR's assumed in the transient analyses are equal to or greater than the established MCPR operating limits for the two fuel types. This results in a conservative ΔCPR and is acceptable.

Therefore, based on conservative analyses, the aforementioned operating limit MCPR's will ensure that the fuel cladding integrity safety limit MCPR of 1.06 will not be violated during the most adverse anticipated transient or fuel loading error that may occur during Cycle 4. Thus,it is concluded that MCPR operating limits of 1.35 and 1.32,at rated conditions,for the 7x7 and 8x8 fuel types respectively are acceptable for Quad Cities Unit No. 1 during Cycle 4.

2.3.3 Rod Withdrawal Error

The rod withdrawal error transient is discussed in Reference 3 for worst case conditions. The event description and analysis assumptions for the rod withdrawal error are given in Reference 4. The information in these references indicates that the local power range monitor subsystem will detect and alarm a high local power condition. However, if the reactor operator ignores the LPRM alarm, the rod block monitor subsystem, set at 107% of full rated power at 100% core flow, will terminate the RWE transient in time to limit the maximum change in the critical power ratio to 0.24 for 7x7 fuel, 0.11 for 8x8 fuel. A RBM rod block occurring at 107% power and full core flow results in a peak linear heat generation rate of 18.0Kw/ft and 16.8 Kw/ft for 7x7 and 8x8 fuel types respectively. These calculated LHGR's are below the 1% plastic strain LHGR's for 7x7 and 8x8 fuels respectively and are acceptable.

The rod withdrawal error analysis is based on the most reactive reactor state and conservatively assumes no xenon, which maximizes the amount of excess reactivity inserted upon withdrawal of the maximum worth control rod from the core. The analysis also assumes the most severe rod block monitor detector in operability allowed by the Technical Specifications.

The RWE ΔCPR is greater than the ΔCPR for all other abnormal operation transients (excluding the fuel loading error) analyzed for 7x7 fuel. The RWE is not the limiting transient however for 8x8 fuel. For the 8x8 fuel the turbine trip without bypass occurring at EOC is the limiting abnormal operational transient. The operating limit MCPR's based on the Cycle 4 fuel loading error is more conservative than the limit associated with operational transients and therefore precludes the localized RWE transient from exceeding the safety limit MCPR of 1.06. We, therefore, conclude that the analysis performed for the rod withdrawal error and the predicted consequences are acceptable.

2.3.4 Turbine Trip Without Bypass

This transient produces the most severe reactor isolation. The reactor pressure increase due to fast closure of the turbine stop valves causes a significant decrease in the core void fraction which in turn induces a positive core reactivity insertion, resulting a rapid and substantial increase in the neutron flux. The transient is terminated by a reactor trip initiated by fast closure position switches on the turbine stop valves.

The transient analyses for Quad Cities Unit No. 1 were performed at exposures corresponding to 1000 Mwd/t before EOC-4 conditions and EOC-4. Since the severity of this transient increases with burnup, the former analysis provides conservative results for reactor operation from BOC-4 to 1000 Mwd/t before EOC-4 while the latter provides conservative results for operation from 1000 Mwd/t before the end of cycle to the end of Cycle 4. The analysis for 1000 Mwd/t before EOC-4 was performed assuming an initial reactor power level of 100% while the EOC-4 analysis assumed the transient was initiated from 98% power. The latter analysis was performed at 98% power because at EOC this represents the maximum power level allowed to maintain at least a 25 psi margin below the set point of the safety valve with the lowest (1240 psig) opening pressure. The analysis results provided in Table 6-2 of Reference 3 show that at EOC a 26 psi margin exists, between the peak transient pressure and the set point of the lowest safety valve while at 1000 Mwd/t before EOC a 27 psi margin is available. This is acceptable to the Staff.

The turbine trip without bypass also results in a significant reduction in the MCPR. This is caused by the rapid and substantial increase in the neutron flux which results in a significant increase in the fuel surface heat flux. The results presented in Table 6-2 of Reference 3 show that the increase in neutron flux and heat flux is more severe for the EOC condition at 98% power than a 1000 Mwd/t before EOC at 100% power. This transient behavior can be attributed to a somewhat less responsive scram reactivity function at the end of cycle. The analysis of this transient also includes design conservatism factors of .80 applied to the scram reactivity curve and 1.25 on the void coefficient. The most severe ΔCPR's for this transient at EOC, were 0.17 and 0.21 for the 7x7 and 8x8 fuel types respectively.

Except for the fuel loading error, the turbine trip without bypass is the most limiting event for the 8x8 fuel during Cycle 4. Similarly, the turbine trip without bypass results in a smaller ΔCPR for the 7x7 fuel than the rod withdrawal error or the fuel loading error. It is concluded, therfore, that operating limit MPCR's for both fuel types based on the limiting Fuel Loading Error will necessarily assure that the transient MCPR's will not violate the safety limit MCPR of 1.06 should a turbine trip without bypass transient occur at Quad Cities Unit No. 1 during Cycle 4. This is acceptable to the Staff.

2.3.5 Operating MCPR Limits for Less than Rated Flow

To assure that the safety limit MCPR is not violated for the limiting flow increase transient (recirculation pump speed control failure) starting from less than rated flow conditions, the licensee will operate Quad Cities No. 1 in conformance with the limiting conditions for operation as stated in paragraph 3.5-K of the Technical Specifications. This requires that for core flow rates less than full rated flow, the licensee shall maintain the MCPR above the minimum operating values. The minimum MCPR values for less than full rated flow are equal to the MCPR for rated flow (1.35 for 7x7, 1.32 for 8x8) multiplied by the respective K_f factor values appearing in Figure 3.5-2 of the Technical Specifications. The K_f factor curves were generically derived and assure that for the most limiting flow increase transients, occurring from less than rated core flow, the actual MCPR will not exceed the safety limit MCPR of 1.06.

We conclude that application of the above stated K_f factors for reduced flow conditions, results in calculated consequences for the limiting anticipated flow increase transients, which do not exceed the thermal limits of the fuel or the pressure limits of the reactor coolant boundary.

Based upon the above, we conclude the the analyses and operating limits based upon the use of the General Electric Thermal Analysis Basis have been conservatively applied to Reload 3 (Cycle 4) and are acceptable.

2.4 Accident Analysis

2.4.1 ECCS Appendix K Analysis

The licensee's evaluation of the performance of the Emergency Core Cooling System (ECCS) for Cycle 4 is contained in Section 6.3.2.4 of Reference 3. This evaluation utilized the General Electric ECCS model [9, 10, 11] that has been previously reviewed and approved by the Staff for application to Quad Cities Station Unit No. 1[13].

Recently, the licensee has informed the Staff that several errors have been discovered in the Quad Cities Unit No. 1 ECCS evaluation and in a number of other evaluations for similar reactors. Upon notification by the licensee, the Staff modified the Quad Cities Operating License[14] in a manner that conservatively corrects these errors and assures that operation of the reactor is in conformance with the performance criteria of 10 CFR 50.46.

The operation of the Quad Cities Unit No. 1 facility is technically not in conformance with the requirments of 10 CFR 50.46 without reevaluation which uses the approved model. However, specific ECCS performance evaluations for the facility, incorporating the revised and approved models with the input errors corrected, will not be completed for some time. Because the Staff has found that the limitations on MAPLHGR, as set forth in Reference 14, will assure that the ECCS will conform to the performance criteria of 50.46 during Cycle 4, the Commission has exempted[14] Commonwealth Edison from the requirements of 10 CFR 50.46 that relate to the use of approved model until a revised ECCS anlaysis for Quad Cities Unit No. 1 can be completed.

2.4.2 Steamline Break Accident

The spectrum of steamline break accidents which are postulated to occur inside containment are covered by the ECCS anlalysis is discussed in Section 2.4.1. The analysis results and conclusions of steamline break accidents occurring outside containment, as presented by the licensee, are acceptable based on the generic staff review of NEDO-20360[4].

2.4.3 Fuel Loading Error

Fuel loading errors are discussed in Reference 3 for a fresh 8x8 fuel bundle placed in core position analyzed for a high burnup 8x8 fuel assembly or rotated 180 degrees in a location near the center of the core. The information in Reference 3 indicates that the most severe fuel loading error results in a peak linear heat generation rate (LHGR) of 16.5 kw/ft and a minimum critical power ratio of 1.01 in the misplaced fuel bundle during steady-state full power operating conditions. Fuel bundles adjacent to a misloaded fuel bundle are negligibly effected. The calculated peak LHGR is less than that which would cause a 1% plastic strain in the cladding. However, the calculated MCPR of 1.01 in the misloaded bundle violates the fuel integrity safety limit MCPR of 1.06.

The fuel loading error is being generically reviewed by the Staff and a generic resolution is anticipated. In the interim, we require that the licensee increase the MCPR operating limit, for all fuel types, to values which will assure that during normal operation the safety limit MCPR will not be violated due to a fuel loading error. An increase of.05 in the 7x7 and 8x8 fuel operating limit MCPR's is sufficient to accomplish this. Thus, for Cycle 4 of Quad Cities Unit No. 1, MCPR operating limits of 1.35 for the 7x7 fuel and 1.32 for the 8x8 fuel will assure that the most severe fuel loading error will not cause a violation of the safety limit MCPR. The licensee has agreed to these MCPR operating limits. The Staff concludes, therefore, that the consequences of a postulated fuel loading error are acceptable.

the appropriate scram, void and Doppler reactivities. No credit for relief valve operation was assumed and all safety valves were assumed operable. The results of the analysis at the cycle exposures analyzed, indicate that the peak pressure at the bottom of the vessel would be 1312 psig for the more sever case. Furthermore, generic analyses (15) applied to Quad Cities Unit No. 1 showed that for this limiting over-pressure transient, the failure of 1 safety valve would cause the maximum vessel pressure to increase by less than 20 psi. Hence the maximum transient pressure, at the bottom of the reactor vessel, caused by fast closure of all MSIV's, with indirect high flux scram, no relief valve actuation and one failed safety valve, results in at least 43 psi margin to the ASME vessel code limit of 1375 psig (110% of 1250 psig). This result is acceptable to the staff.

2.6 Thermal-Hydraulic Stability Analysis

A Cycle 4 thermal-hydraulic stability analysis, using the analytical methods discussed in Reference 4, was presented by the licensee for Quad Cities Unit No. 1.

The results of the Cycle 4 analysis show that the 7x7 and 8x8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is well within the channel operational design guide decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios are also both within the reactor core operational design guide decay ratio. These results are acceptable to the Staff.

The NRC Staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve change. The Staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a stability test program.

The stability testing has recently been completed by GE and the licensee of a large BWR-4 nuclear power plant. Although a test report has not yet been received by the Staff, it is expected that the test results will provide considerable aid in resolving the Staff concerns.

In the interim the Staff has imposed a requirement on Quad Cities Unit No. 1 which will restrict planned operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. This restriction will provide a significant increase in the reactor core stability margins at Quad Cities Unit No. 1 during Cycle 4. On the basis of the foregoing, the NRC Staff considers the thermal-hydraulic stability of Quad Cities Unit No. 1 to be acceptable.

2.4.4 Control Rod Drop Accident

The Cycle 4 control rod drop accident for Quad Cities Unit No. 1 is not within the generic bounding analysis presented in Reference 4. Although the actual Cycle 4 Doppler coefficients and accident reactivity shape functions for the cold and hot startup conditions conservatively fall within the values assumed in the bounding analysis, the scram reactivity shape functions for both hot and cold startup conditions do not. Therefore, the licensee has performed a plant specific control rod drop accident analysis for Quad Cities Unit No. 1 for Cycle 4.

The plant specific analysis was performed using actual hot and cold Doppler coefficients of reactivity corresponding to the beginning of cycle, which represents the most limiting cycle exposure for this accident, since the Doppler coefficient is least negative at the beginning of the cycle.

The results of the analysis show that the positive reactivity insertion rate of the dropped rod is compenstated sufficiently by Doppler feedback and scram reactivity effects, to limit the energy deposition in the fuel to a maximum of 180.5 cal/gm for cold (20^{o}C) startup and 231.8 cal/gm for hot (286^{o}C) startup conditions.

Thus, we have concluded that the results of a control rod drop accident from any in-sequence control rod movement will be below the design limit of 280 cal/gm and therefore has acceptable consequences.

2.4.5 Fuel Handling Accident

The licensee notes in Reference 3 that fuel handling accident description, analysis and results provided in the FSAR and discussed in the generic reload topical report[4] are applicable to the 8x8 reload fuel. That is, the total activity released to the environment and the resulting radiological exposures for the reload fuel will be less than those values presented in the FSAR for the 7x7 core. As identified in the FSAR, the radiological exposures for this accident with 7x7 fuel are well below the guidelines set forth in 10 CFR Part 100. Therefore, we have concluded that the consequences of this accident for the 8x8 fuel will also be well below the 10 CFR 100 guidelines.

2.5 Overpressure Analysis

The licensee presented the results of an overpressure analysis[1] to demonstrate that an adequate margin exists to the ASME code allowable vessel pressure, which is 110% of the vessel design pressure. The transient analyzed was the fast closure of all main steamline isolation valves with the conservative assumption that a reactor scram would occur on the second (high neutron flux) scram signal rather than the first (10% valve closure position switches). The licensee analyzed this event at two Cycle 4 exposures corresponding to 1000 Mwd/t before EOC-4 and EOC. The analysis was performed at these burnups near and at EOC since the nuclear parameters tend to make the consequences more limiting toward the end of cycle. The analysis was performed for 98% and 100% of licensed power with the appropriate scram, void and Doppler reactivities. No credit for relief valve operation was assumed and all safety valves were assumed operable. The results of the analysis at the cycle exposures analyzed, indicate that the peak pressure at the bottom of the vessel would be (15) 1312 psig for the more sever case. Furthermore, generic analyses applied to Quad Cities Unit No. 1 showed that for this limiting overpressure transient, the failure of 1 safety valve would cause the maximum vessel pressure to increase by less than 20 psi. Hence the maximum transient pressure, at the bottom of the reactor vessel, caused by fast closure of all MSIV's, with indirect high flux scram, no relief valve actuation and one failed safety valve, results in at least 43 psi margin to the ASME vessel code limit of 1375 psig (110% of 1250 psig). This result is acceptable to the staff.

2.6 Thermal-Hydraulic Stability Analysis

A Cycle 4 thermal-hydraulic stability analysis, using the analytical methods discussed in Reference 4, was presented by the licensee for Quad Cities Unit No. 1.

The results of the Cycle 4 analysis show that the 7x7 and 8x8 channel hydrodynamic stability, at either rated power and flow conditions or at the low end of the flow control range, is well within the channel operational design guide decay ratio. Calculations were also performed by the licensee to assess the reactor power dynamic response at the two aforementioned reactor operating conditions. The results of this analysis showed that the reactor core decay ratios are also both within the reactor core operational design guide decay ratio. These results are acceptable to the Staff.

The NRC Staff has expressed generic concerns regarding reactor core thermal-hydraulic stability at the least stable reactor condition allowed by Technical Specifications. This condition could be reached during an operational transient from high power where the plant sustains a trip of both recirculation pumps without a reactor trip. The concerns are motivated by increasing decay ratios as equilibrium fuel cycles are approached and as fuel designs improve change. The Staff concerns relate to both the consequences of operating at a decay ratio of 1.0 and the capacity of analytical methods to accurately predict decay ratios. The General Electric Company is addressing the staff concerns through meetings, topical reports and a stability test program.

The stability testing has recently been completed by GE and the licensee of a large BWR-4 nuclear power plant. Although a test report has not yet been received by the Staff, it is expected that the test results will provide considerable aid in resolving the Staff concerns.

In the interim the Staff has imposed a requirement on Quad Cities Unit No. 1 which will restrict planned operations in the natural circulation flow mode. The licensee has agreed to this Technical Specification limitation. This restriction will provide a significant increase in the reactor core stability margins at Quad Cities Unit No. 1 during Cycle 4. On the basis of the foregoing, the NRC Staff considers the thermal-hydraulic stability of Quad Cities Unit No. 1 to be acceptable.

3.0 Physics Startup Testing

As part of the review of Reload 3 of Quad Cities Unit No. 1 the licensee was requested to provide a description of the Cycle 4 physics startup test program and a schedule for reporting of the test results. In response to that request, a proposed physics startup test program was provided by the licensee. The staff finds that the combined physics startup tests, along with the tests required to assure compliance with the Technical Specifications, provide an acceptable physics startup test program.

The results of these tests will be available for inspection and review at the station within 90 days of the completion of the test program. This is also acceptable to the Staff.

4.0 Technical Specification Changes

The proposed Technical Specifications for Cycle 4 operation of Quad Cities Unit No. 1, include a change in the MCPR operating limits for the 7x7 and 8x8 fuel types, based on the safety analyses results presented in Reference 3. As discussed in this evaluation, the proposed operating limit MCPR's must be raised to insure that the fuel cladding integrity safety limit is not violated in the event of a fuel loading error. The required Technical Specification MCPR operating limits for each fuel type conform to the results of the fuel loading error analysis as presented in Reference 3.

The licensee did not propose, as part of the Reload 3 licensing submittal, any changes to the MAPLHGR curves contained in the Quad Cities Unit No. 1 Technical Specifications. Subsequent to the reload submittal, however, the licensee voluntarily instituted a reduction in the Technical Specification MAPLHGR limits to account for the GE ECCS input errors discussed in this evaluation. The Quad Cities Operating License has been modified by Order of the Director of the Office of Nuclear Reactor Regulation to include these voluntary reductions until Commonwealth Edison submits a reevaluation of the Quad Cities ECCS performance using an approved model. These revised fuel type dependent MAPLHGR curves are acceptable during Cycle 4 operation of Quad Cities Unit No. 1.

In order to provide a significant increase in the reactor core stability margins during Cycle 4 the licensee has agreed to Technical Specifications which restrict all planned reactor operation in natural circulation.

ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

CONCLUSION

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 3, 1977

REFERENCES

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

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 Reload No. 3 Licensing Submittal for Quad Cities Nuclear Power Station Unit 1." NEDO-21489, submitted as Enclosure A of Commonwealth Edison's January 21, 1977 Proposed Amendment to License DPR-29.

4. "General Electric Generic Reload LIcensing Application for 8x8 Fuel," Revision 1, Supplement 4, April, 1976, NEDO-20360.

5. Status Report on the Licensing Topical Report "General Electric Boiling Water Reactor Generic Reload Application for 8x8 Fuel", NEDO-20360 Revision 1 and Supplement 1 by Division of Technical Review, Office of Nuclear Reactor Regulation, US Nuclear Regulatory Commission, April, 1975.

6. "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application", General Electric Company, BWR Systems Department, November, 1973 (NEDO-10958).

7. "Topical Report Evaluation of General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application", by Division of Technical Review, Office of Nuclear Reactor Regulation, US Nuclear Regulatory Commission, September, 1974.

8. General Electric Letter (John Hinds) to US Atomic Energy Commission, (Walter Butler) "Responses to the Third Set of AEC Questions on the General Electric Licensing Topical Reports, NEDO-10958 and NEDO-19058, "General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application." July, 1974.

9. Commonwealth Edison letter (Abel) to NRC (Ziemann), "Quad Cities Station Unit 2, Special Report No. 15, Supplement C, NRC Docket No. 50-265", dated April 18, 1975.

10. Commonwealth Edison letter (Abel) to NRC (Ziemann) "Quad Cities Station Unit 2, Special Report No. 15, Supplement C, NRC Docket No. 50-265" (revision) - dated April 21, 1975.

11. Commonwealth Edison letter (Bolger) to NRC (Rusche), "Quad Cities Station Unit 1 Proposed Amendment to Facility Operating License No. DPR-29 and Quad Cities Station Special Report No. 15, Supplement D, NRC Docket No. 50-254", dated July 1, 1975.

12. Safety Evaluation Report of Quad Cities Nuclear Power Station Unit 1 and 2, Docket Nos. 50-254, and 50-265, issued December 27, 1974.

13. USNRC Letter (K. Goller) to General Electric (G. Sherwood) transmitting "Safety Evaluation for General Electric ECCS Evaluation Model Modifications", dated April 12, 1977.

14. USNRC Letter (Ziemann) to Commonwealth Edison (Bolger) dated March 11, 1977 transmitting the Commission's Order for Modification of License and Exemption.

15. General Electric Letter (Ivan F. Stuart) to US Nuclear Regulatory Commission (Victor Stello) "Code Overpressure Protection Analysis Sensitivity of Peak Vessel Pressures to Valve Operability," December 23, 1975.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-254

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 41 to Facility Operating License No. DPR-29, issued to Commonwealth Edison Company (acting for itself and on behalf of the Iowa-Illinois Gas and Electric Company), which revised the license and Technical Specifications appended thereto for operation of the Quad Cities Nuclear Power Station Unit No. 1 (the facility) located in Rock Island County, Illinois. The amendment is effective as of its date of issuance.

The amendment (1) authorized operation with additional 8 x 8 fuel assemblies, (2) incorporated revised MAPLHGR and MCPR limits in response to the plant specific analysis for reload 3, and (3) modified License Condition 3.C to reflect end-of-cycle scram reactivity conditions for reload 3.

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

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated January 21, 1977, and a supplement thereto dated April 25, 1977, (2) Amendment No. 41 to License No. DPR-29, and (3) the Commission's 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, 504 - 17th Street, Moline, Illinois 60625. A 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 Operating Reactors.

Dated at Bethesda, Maryland, this third day of May, 1977.

FOR THE NUCLEAR REGULATORY COMMISSION

Don K. Davis

Don K. Davis, Acting Chief
Operating Reactors Branch #2
Division of Operating Reactors