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Docket No. 50-265

December 23, 1981

Mr. L. DelGeorge
Director of Nuclear Licensing
Commonwealth Edison Company
P. O. Box 767
Chicago, Illinois 60690

Dear Mr. DelGeorge:

The Commission has issued the enclosed Amendment No. 69 to Facility Operating License No. DPR-30 for Quad Cities Station Unit 2. This amendment is in response to your request dated July 27, 1981, as supplemented by letters dated August 21, 1981, and December 3, 1981.

This amendment (1) authorizes operation in Cycle 6 using 224 assemblies of prepressurized 8 x 8R fuel, including 144 bundles of GE barrier fuel, (2) incorporates revised Minimum Critical Power Ratio (MCPR) limits in response to plant specific analyses for Cycle 6, (3) incorporates new Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the barrier fuel, (4) deletes MCPR, MAPLHGR and Linear Heat Generation Rate (LHGR) operating limits for all 7 x 7 fuel (none to remain in the core), and (5) changes the pressure safety limits due to the recently installed Anticipated Transients Without Scram Recirculation Pump Trip.

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

Sincerely,

ORIGINAL SIGNED BY

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 69 to DPR-30
2. Safety Evaluation
3. Notice

cc w/enclosures
See next page

Goddard amdt + notice only

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DATE	12/18/81	12/21/81	12/21/81	12/22/81	12/22/81	

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES STATION UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENCE

Amendment No. 69
License No. DPR-30

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

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated July 27, 1981, as supplemented August 21, 1981 and December 3, 1981, 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 Commisison;

   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 paragraph 3.B of Facility Operating License No. DPR-30 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 69 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: December 23, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 69

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

1.1/2.1-4
1.1/2.1-5
1.1/2.1-6
1.1/2.1-7
1.1/2.1-7a (new page)
1.1/2.1-11
1.2/2.2-1
1.2/2.2-2
1.2/2.2-2a (new page)
3.3/4.3-5
3.3./4.3-10
3.5/4.5-10
3.5/4.5-14
3.5/4.5-14a
3.5/4.5-14b
Figure 3.5-1 (sheet 1 of 5)
Figure 3.5-1 (sheet 2 of 5)
Figure 3.5-1 (sheet 4 of 5)

1.1 SAFETY LIMIT BASIS

The fuel cladding integrity limit is set such that no calculated fuel damage would occur as a result of an abnormal operational transient. Because fuel damage is not directly observable, a step-back approach is used to establish a safety limit such that the minimum critical power ratio (MCPR) is no less than the fuel cladding integrity safety limit. MCPR > the fuel cladding integrity safety limit represents a conservative margin relative to the conditions required to maintain fuel cladding integrity.

The fuel cladding is one of the physical boundaries which separate radioactive materials from the environs. The integrity of the fuel cladding is related to its relative freedom from perforations or cracking. Although some corrosion or use-related cracking may occur during the life of the cladding, fission product migration from this source is incrementally cumulative and continuously measurable. Fuel cladding perforations, however, can result from thermal stresses which occur from reactor operation significantly above design conditions and the protection system safety settings. While fission product migration from cladding perforation is just as measurable as that from use-related cracking, the thermally caused cladding perforations signal a threshold beyond which still greater thermal stresses may cause gross rather than incremental cladding deterioration. Therefore, the fuel cladding safety limit is defined with margin to the conditions which would produce onset of transition boiling (MCPR of 1.0). These conditions represent a significant departure from the condition intended by design for planned operation. Therefore, the fuel cladding integrity safety limit is established such that no calculated fuel damage shall result from an abnormal operational transient. Basis of the values derived for this safety limit for each fuel type is documented in References 1 and 2.

A. Reactor Pressure > 800 psig and Core Flow > 10% of Rated

Onset of transition boiling results in a decrease in heat transfer from the cladding and therefore elevated cladding temperature and the possibility of cladding failure. However, the existence of critical power, or boiling transition is not a directly observable parameter in an operating reactor. Therefore, the margin to boiling transition is calculated from plant operating parameters such as core power, core flow, feedwater temperature, and core power distribution. The margin for each fuel assembly is characterized by the critical power ratio (CPR), which is the ratio of the bundle power which would produce onset of transition boiling divided by the actual bundle power. The minimum value of this ratio for any bundle in the core is the minimum critical power ratio (MCPR). It is assumed that the plant operation is controlled to the nominal protective setponts via the instrumented variables (Figure 2.1-3).

The MCPR fuel cladding integrity safety limit has sufficient conservatism to assure that in the event of an abnormal operational transient initiated from the normal operating condition, more than 99.9% of the fuel rods in the core are expected to avoid boiling transition. The margin between MCPR of 1.0 (onset of transition boiling) and the safety limit, is derived from a detailed statistical analysis considering all of the uncertainties in monitoring the core operating state, including uncertainty in the boiling transition correlation (see e.g., Reference 1). Because the boiling transition correlation is based on a large quantity of full-scale data, there is a very high confidence that operation of a fuel assembly at the condition of MCPR = the fuel cladding integrity safety limit would not produce boiling transition.

However, if boiling transition were to occur, cladding perforation would not be expected. Cladding temperatures would increase to approximately 1100°F, which is below the perforation temperature of the cladding material. This has been verified by tests in the General Electric Test Reactor (GETR), where similar fuel operated above the critical heat flux for a significant period of time (30 minutes) without cladding perforation.

If reactor pressure should ever exceed 1400 psia during normal power operation (the limit of applicability of the boiling transition correlation), it would be assumed that the fuel cladding integrity safety limit has been violated.

In addition to the boiling transition limit (MCPR) operation is constrained to a maximum LHGR ≤ 17.5 kw/ft for 7 x 7 fuel and 13.4kw/ft for all 8x8 fuel types. This constraint is established by Specification 3.5.J. to provide adequate safety margin to 1% plastic strain for abnormal operating transients initiated from high power conditions. Specification 2.1.A.1 provides for equivalent safety margin for transients initiated from lower power conditions by adjusting the APRM flow-biased scram setting by the ratio of FRP/MFLPD.

Specification 3.5.J established the LHGR maximum which cannot be exceeded under steady power operation.

B. Core Thermal Power Limit (Reactor Pressure<800 psia)

At pressures below 800 psia, the core elevation pressure drop (0 power, 0 flow) is greater than 4.56 psi. At low powers and flows this pressure differential is maintained in the bypass region of the core. Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low powers and flows will always be greater than 4.56 psi. Analyses show that with a flow of 28 x 10^3 lb/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus the bundle flow with a 4.56-psi driving head will be greater than 28 x 10^3 lb/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. At 25% of rated thermal power, the peak powered bundle would have to be operating at 3.86 times the average powered bundle in order to achieve this bundle power. Thus, a core thermal power limit of 25% for reactor pressures below 800 psia is conservative.

C. Power Transient

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 to 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. 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. Control rod scram times are checked as required by Specification 4.3.C. and the MCPR operating limit is modified as necessary per Specification 3.5.K.

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 MCPR = the fuel cladding integrity safety limit is not exceeded. Thus, use of a 1.5 second limit provides additional margin.

The computer provided 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 requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core-cooling capability could lead to elevated cladding temperatures and cladding perforation. The core will be cooled sufficiently to prevent cladding melting should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel* provides adequate margin. This level will be continuously monitored whenever the recirculation pumps are not operating.

*Top of the active fuel is defined to be 360 inches above vessel zero (see Bases 3.2).

References

1. "Generic Reload Fuel Applications," NEDE-24011-P-A*

2. "Generic Information For Barrier Fuel Demonstration Bundle Licensing", NEDO-24259-A, February 1981.

*Approved revision number at time reload fuel analyses are performed.

2.1 LIMITING SAFETY SYSTEM SETTING BASES

The abnormal operational transients applicable to operation of the units have been analyzed throughout the spectrum of planned operating conditions up to the rated thermal power condition of 2511 MWt. In addition, 2511 MWt is the licensed maximum steady-state power level of the units. This maximum steady-state power level will never knowingly be exceeded.

Conservatism incorporated into the transient analysis is documented in References 1 and 2. Transient analyses are initiated at the conditions given in these References.

The scram delay time and rate of rod insertion allowed by the analyses are conservatively set equal to the longest delay and slowest insertion rate acceptable by technical specifications. The effects of scram worth, scram delay time, and rod insertion rate, all conservatively applied, are of greatest significance in the early portion of the negative reactivity insertion. The rapid insertion of negative reactivity is assured by the time requirements for 5% and 20% insertion. By the time the rods are 60% inserted, approximately 4 dollars of negative reactivity have been inserted, which strongly turns the transient and accomplishes the desired effect. The times for 50% and 90% insertion are given to assure proper completion of the expected performance in the earlier portion of the transient, and to establish the ultimate fully shutdown steady-state condition.

The MCPR operating limit is, however, adjusted to account for the statistical variation of measured scram times as discussed in Reference 2 and the bases of Specification 3.5.K.

Steady-state operation without forced recirculation will not be permitted except during startup testing. The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps.

The bases for individual trip settings are discussed in the following paragraphs.

For analyses of the thermal consequences of the transients, the MCPR's stated in Paragraph 3.5.K as the limiting condition of operation bound those which are conservatively assumed to exist prior to initiation of the transients.

A. Neutron Flux Trip Settings

1. APRM Flux Scram Trip Setting (Run Mode)

The average power range monitoring (APRM) system, which is calibrated using heat balance data taken during steady-state conditions, reads in percent of rated thermal power. Because fission chambers provide the basis input signals, the APRM system responds directly to average neutron flux. During transients, the instantaneous rate of heat transfer from the fuel (reactor thermal power) is less than the instantaneous neutron flux due to the time constant of the fuel.

Therefore, during abnormal operational transients, the thermal power of the fuel will be less than that indicated by the neutron flux at the scram setting. Analyses demonstrate that with a 120% scram trip setting, none of the abnormal operational transients analyzed violates the fuel safety limit, and there is a substantial margin from fuel damage. Therefore, the use of flow-referenced scram trip provides even additional margin.

References

1. "Generic Reload Fuel Application," NEDE-24011-P-A*

*Approved revision number at time reload analyses are performed

2. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" General Electric Co. Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Volume III as supplemented by letter dated September 5, 1980 from R. H. Buchholz (GE) to P. S. Check (NRC).

1.2/2.1 REACTOR COOLANT SYSTEM

SAFETY LIMIT

Applicability:

Applies to limits on reactor coolant system pressure.

Objective:

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

LIMITING SAFETY SYSTEM SETTING

Applicability:

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

Objective:

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

SPECIFICATIONS

A. The reactor coolant system pressure as measured by the vessel steam space pressure indicator shall not exceed 1345 psig at any time when irradiated fuel is present in the reactor vessel.

A. Reactor coolant high-pressure scram shall be ≤1060 psig.

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

1 valve at 1115psig(1)
2 valves at 1240 psig
2 valves at 1250 psig
4 valves at 1260 psig

(1)Target Rock combination safety/relief valve

The allowable setpoint error for each valve shall be ±1%.

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 1345 psig as measured by the vessel steam space pressure indicator is equivalent to 1375 psig at the lowest elevation of the reactor vessel. 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 design pressure (120% x 1175 = 1410 psig). The safety limit pressure of 1375 psig is referenced to the lowest elevation of the reactor vessel. The design pressure for the recirc. suction line piping (1175 psig) was chosen relative to the reactor vessel design pressure. Demonstrating compliance of peak vessel pressure with the ASME overpressure protection limit (1375 psig) assures compliance of the suction piping with the USASI limit (1410 psig). Evaluation methodology to assure that this safety limit pressure is not exceeded for any reload is documented in Reference 1. 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 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 pressure to approximately 1100 psig (References 2,3, and 4). In addition, pressure relief valves have been provided to reduce the probability of the safety valves operating in the event that the turbine bypass should fail.

Finally, the safety valves are sized to keep the reactor vessel peak 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 indirect flux scram and safety valve actuation provide adequate margin below the allowable peak 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. "Generic Reload Fuel Application," NEDE-24011-P-A*

2. SAR, Section 11.22

3. Quad Cities 1 Nuclear Power Station first reload license submittal, Section 6.2.4.2, February 1974.

4. GE Topical Report NEDO-20693, General Electric Boiling Water Reactor No. 1 Licensing submittal for Quad Cities Nuclear Power Station Unit 2, December 1974.

* Approved revision number at time reload analyses are performed.

sidered inoperable, fully inserted into the core, and electrically disarmed.

5. If the overall average of the 20% insertion scram time data generated to date in the current cycle exceeds 0.73 seconds, the MCPR operating limit must be modified as required by Specification 3.5.K.

provide reasonable assurance that proper control rod drive performance is being maintained. The results of measurements performed on the control rod drives shall be submitted in the annual operating report to the NRC.

5. The cycle cumulative mean scram time for 20% insertion will be determined immediately following the testing required in Specifications 4.3.C.1 and 4.3.C.2 and the MCPR operating limit adjusted, if necessary, as required by Specification 3.5.K.

D. Control Rod Accumulators

At all reactor operating pressures, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has:

1. an inoperable accumulator,
2. a directional control valve electrically disarmed while in a nonfully inserted position, or
3. a scram insertion greater than maximum permissible insertion time.

If a control rod with an inoperable accumulator is inserted full-in and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator, and the rod block associated with that inoperable accumulator may be bypassed.

D. Control Rod Accumulators

Once a shift, check the status of the pressure and level alarms for each accumulator.

E. Reactivity Anomalies

The reactivity equivalent of the difference between the actual critical rod configuration and the expected configuration during power operation shall not exceed 1%Δk. If this limit is exceeded, the reactor shall be shutdown until the cause has been determined and corrective actions have been taken. In accordance with Specification 6.6, the NRC shall be notified of this reportable occurrence within 24 hours.

E. Reactivity Anomalies

During the startup test program and startups following refueling outages, the critical rod configurations will be compared to the expected configurations at selected operating conditions. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the critical rod configuration will be compared to the configuration expected based upon appropriately corrected past data. This comparison will be made at least every equivalent full power month.

F. Economic Generation Control System

Operation of the unit with the economic generation control system with automatic flow control shall be permissible only in the range of 65% to 100% of rated core flow, with reactor power above 20%.

F. Economic Generation Control System

The range set into the economic generation control system shall be recorded weekly.

C. Scram Insertion Times

The control rod system is analyzed to bring the reactor subcritical at a rate fast enough to prevent fuel damage, i.e., to prevent the MCPR from becoming less than the fuel cladding integrity safety limit.

Analysis of the limiting power transient shows that the negative reactivity rates resulting from the scram with the average response of all the drives as given in the above specification, provide the required protection, and MCPR remains greater than the fuel cladding integrity safety limit. It is necessary to raise the MCPR operating limit (per Specification 3.5.K) when the average 20% scram insertion time reaches 0.73 seconds on a cycle cumulative basis (overall average of surveillance data to date) in order to comply with assumptions in the implementation procedure for the ODYN transient analysis computer code. The basis for choosing 0.73 seconds is discussed further in the bases for Specification 3.5.K. In the analytical treatment of the transients, 290 milliseconds are allowed between a neutron sensor reaching the scram point and the start of motion of the control rods. This is adequate and conservative when compared to the typically observed time delay of about 210 milliseconds. Approximately 90 milliseconds after neutron flux reaches the trip point, the pilot scram valve solenoid deenergizes and 120 milliseconds later the control rod motion is estimated to actually begin. However, 200 milliseconds rather than 120 milliseconds is conservatively assumed for this time interval in the transient analyses and is also included in the allowable scram insertion times specified in Specification 3.3.C.

The scram times for all control rods will be determined at the time of each refueling outage. A representative sample of control rods will be scram tested following a shutdown.

Scram times of new drives are approximately 2.5 to 3 seconds; lower rates of change in scram times following initial plant operation at power are expected. The test schedule provides reasonable assurance of detection of slow drives before system deterioration beyond the limits of Specification 3.3.C. The program was developed on the basis of the statistical approach outlined below and judgment.

The history of drive performance accumulated to date indicates that the 90% insertion times of new and overhauled drives approximate a normal distribution about the mean which tends to become skewed toward longer scram times as operating time is accumulated. The probability of a drive not exceeding the mean 90% insertion time by 0.75 seconds is greater than 0.999 for a normal distribution. The measurement of the scram performance of the drives surrounding a drive exceeding the expected range of scram performance will detect local variations and also provide assurance that local scram time limits are not exceeded. Continued monitoring of other drives exceeding the expected range of scram times provides surveillance of possible anomalous performance.

The numerical values assigned to the predicted scram performance are based on the analysis of the Dresden 2 startup data and of data from other BWR's such as Nine Mile Point and Oyster Creek.

The occurrence of scram times within the limits, but significantly longer than average, should be viewed as an indication of a systematic problem with control rod drives, especially if the number of drives exhibiting such scram times exceeds eight, the allowable number of inoperable rods.

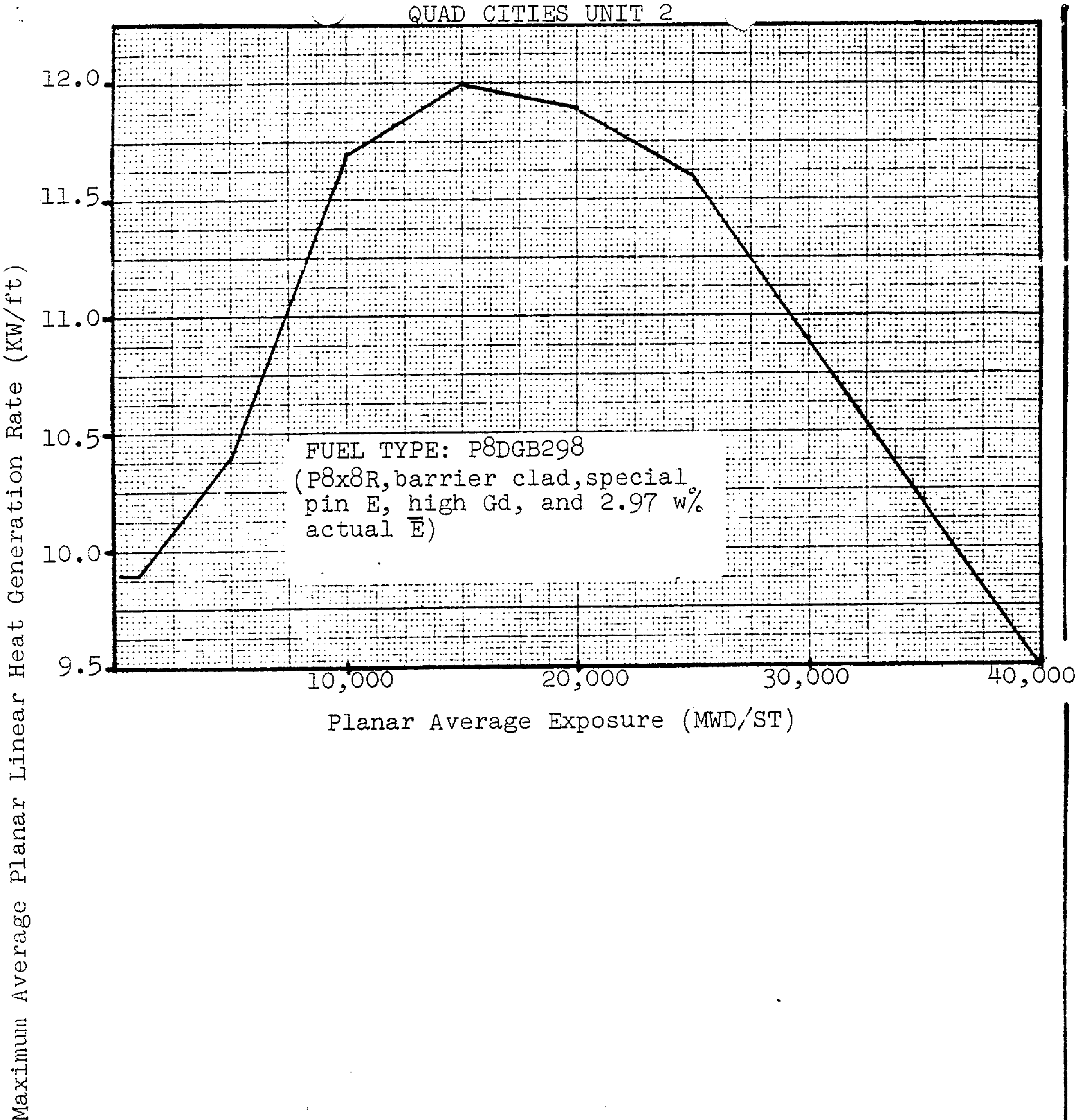

Figure 3.5-1
(Sheet 1 of 5)

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

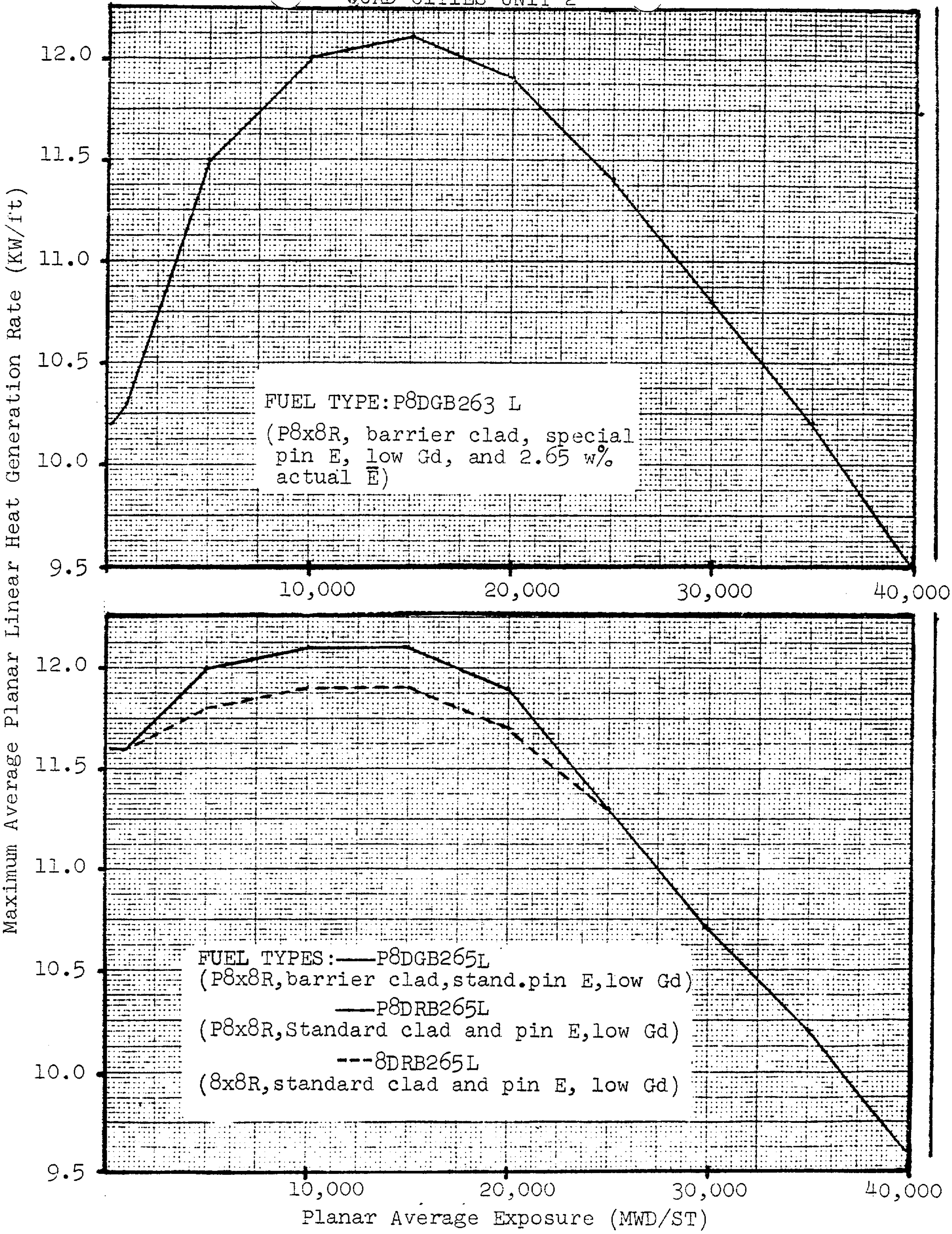

Figure 3.5-1 (Sheet 2 of 5) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

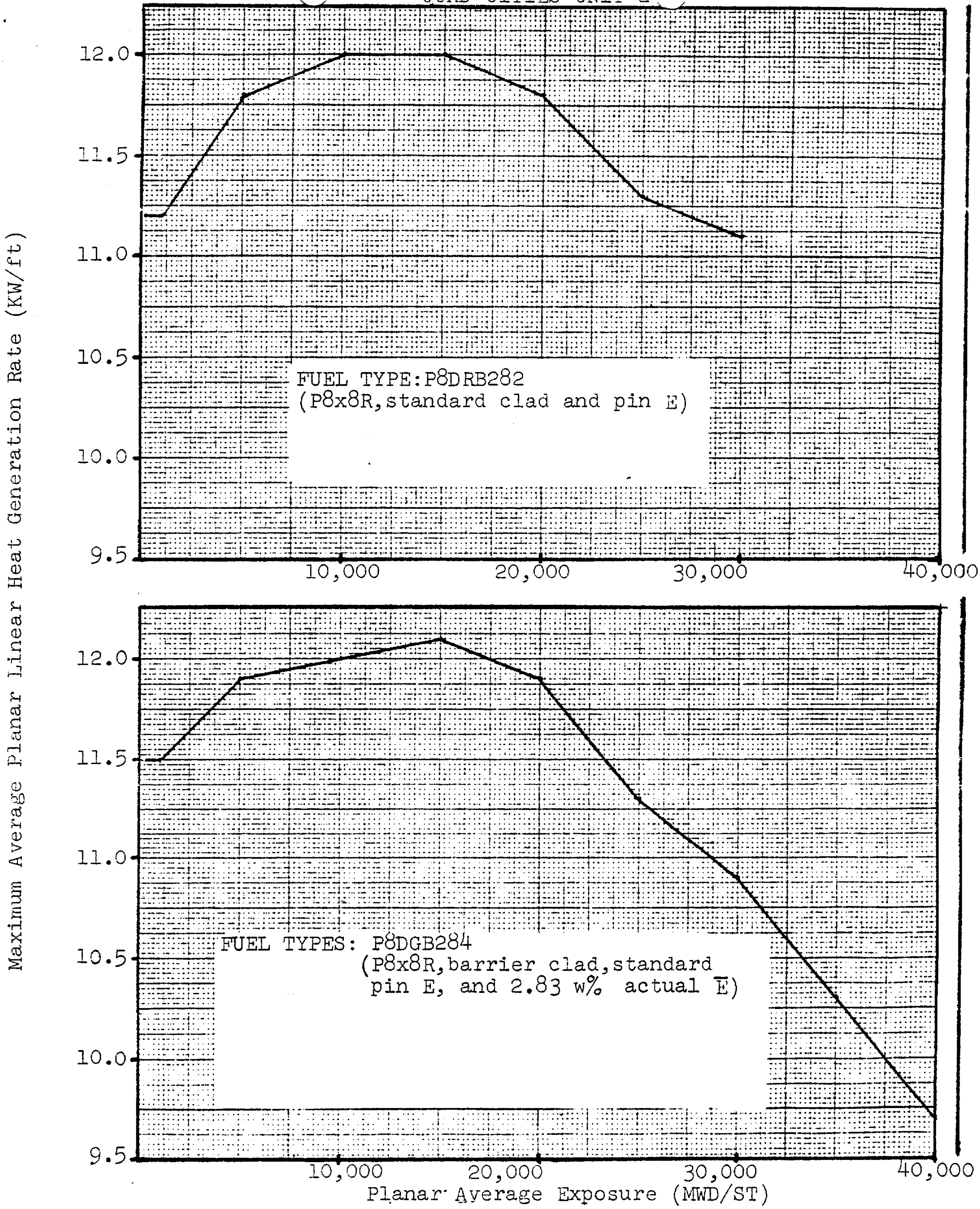

Figure 3.5-1 (Sheet 4 of 5) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

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.

Maximum allowable LHGR for all 8X8 fuel types is 13.4 KW/ft.

K. Minimum Critical Power Ratio (MCPR)

During steady-state operation at rated core flow, MCPR shall be greater than or equal to:

1.37 for $\tau_{ave} \leq 0.73$ secs

1.42 for $\tau_{ave} \geq 0.86$ secs

$0.385\ \tau_{ave} + 1.089$
for $0.73 < \tau_{ave} < 0.86$ secs

where τ_{ave} = mean 20% scram insertion time for all surveillance data from Specification 4.3.C. which has been generated in the current cycle.

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. If 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.

K. Minimum Critical Power Ratio (MCPR)

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

shown on Figure 3.5-1 as limits because conformance calculations have not been performed to justify operation at LHGR's in excess of those shown.

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 is discussed in Reference 2 and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with 95% confidence that no more than one fuel rod exceeds the design LHGR due to power spiking. No penalty is required in Specification 3.5.L because it has been accounted for in the reload transient analyses by increasing the calculated peak LHGR by 2.2%.

K. Minimum Critical Power Ratio (MCPR)

The steady state values for MCPR specified in this specification were selected to provide margin to accomodate 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 intitial condition assumed for the LOCA analysis plus two percent for uncertainity 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 value of MCPR stated in this specification for the limiting condition of operation bounds the initial value of MCPR assumed to exist prior to the initiation of the transients. This initial condition, which is used in the transient analyses, will preclude violation of the fuel cladding integrity safety limit. Assumptions and methods used in calculating the required steady state MCPR limit for each reload cycle are documented in References 2, 4, and 5. 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) Load rejection or turbine trip without bypass

c) Loss of feedwater heater

Several factors influence which of the these transients results in the largest reduction in critical power ratio such as the specific fuel loading, exposure, and fuel type. The current cycle's reload licensing analyses 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 to bound the most restrictive over the entire cycle for each fuel type.

The need to adjust the MCPR operating limit as a function of scram time arises from the statistical approach used in the implementation of the ODYN computer code for analyzing rapid pressurization events. Generic statistical analyses were performed for plant groupings of similar design which considered the statistical variation in several parameters (initial power level, CRD scram insertion time, and model uncertainty). These analyses (which are described further in Reference 4) produced generic Statistical Adjustment Factors which have been applied to plant and cycle specific ODYN results to yield operating limits which provide a 95% probability with 95% confidence that the limiting pressurization event will not cause MCPR to fall below the fuel cladding integrity safety limit.

As a result of this 95/95 approach, the average 20% insertion scram time must be monitored to assure compliance with the assumed statistical distribution. If the mean value on a cycle cumulative (running average) basis were to exceed a 5% significance level compared to the distribution assumed in the ODYN statistical analyses, the MCPR limit must be increased linearly (as a function of the mean 20% scram time) to a more conservative value which reflects an NRC determined uncertainty penalty of 4.4%. This penalty is applied to the plant specific ODYN results (i.e. without statistical adjustment) for the limiting single failure pressurization event occurring at the limiting point in the cycle. It is not applied in full until the mean of all current cycle 20% scram times reaches the 0.90 secs value of Specification 3.3.C.1. In practice, however, the requirements of 3.3.C.1 would most likely be reached (i.e. individual data set average > .90 secs) and the required actions taken (3.3.C.2) well before the running average exceeds 0.90 secs.

The 5% significance level is defined in Reference 4 as:

$$\tau_B = \mu + 1.65 \left(N_1 / \sum_{i=1}^{n} N_i\right)^{1/2} \sigma$$

where μ = mean value for statistical scram time distribution to 20% inserted

σ = standard deviation of above distribution

N_1 = number of rods tested at BOC (all operable rods)

$\sum_{i=1}^{n} N_i$ = total number of operable rods tested in the current cycle

The value for τ_B used in Specification 3.5.K is 0.73 secs which is conservative for the following reasons:

a) For simplicity in formulating and implementing the LCO, a conservative value for $\sum_{i=1}^{n} N_i$ of 708 (i.e. 4x177) was used. This represents one full core data set at BOC plus 6 half core data sets. At the maximum frequency allowed by Specification 4.3.C.2 (16 week intervals) this is equivalent to 24 operating months. That is, a cycle length was assumed which is longer than any past or contemplated refueling interval and the number of rods tested was maximized in order to simplify and conservatively reduce the criteria for the scram time at which MCPR penalization is necessary.

b) The values of μ and σ were also chosen conservatively based on the dropout of the position 39 RPIS switch, since pos. 38.4 is the precise point at which 20% insertion is reached. As a result Specification 3.5.K initiates the linear MCPR penalty at a slightly lower value τ_{ave}. This also produces the full 4.4% penalty at 0.86 secs which would occur sooner than the required value of 0.90 secs.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This ensures 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. "Loss-of-Coolant Analysis Report for Dresden Units 2, 3, and Quad Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A*, April, 1979

2. "Generic Reload Fuel Application," NEDE-24011-P-A**

3. I. M. Jacobs and P. W. Marriott, GE Topical Report APED 5736, "Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards," April, 1969.

4. "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" General Electric Co. Licensing Topical Report NEDO 24154 Vols. I and II and NEDE-24154 Vol. III as supplemented by letter dated September 5, 1980 from R. H. Buchholz (GE) to P. S. Check (NRC).

5. Letter, R. H. Buchholz (GE) to P. S. Check (NRC) dated January 19, 1981 "ODYN Adjustment Methods For Determination of Operating Limits".

* Approved revision at time of plant operation.

** Approved revision number at time reload fuel analyses are performed.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 69 TO FACILITY LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION UNIT NO. 2

DOCKET NO. 50-265

1.0 INTRODUCTION

By letter dated July 27, 1981 the licensee, Commonwealth Edison Company (CECo), proposed changes to the Technical Specifications (TSs) for Quad Cities Unit 2 (see reference 1). These changes are required to support future reloads for Quad Cities Unit 2 in accordance with the provisions of 10 CFR 50.59 and because the barrier fuel demonstration incorporates features not previously addressed and because of the initial application of the ODYN transient analysis code to the upcoming operating cycle. Also, in support of the reload application, the licensee provided a supplemental reload submittal for Quad Cities Unit 2 Reload 5 (Cycle 6) dated August 21, 1981 (see reference 2).

For Reload 5, Cycle 6, 80 bundles of prepressurized General Electric (GE) 8x8 retrofit fuel (P8x8R) and 144 bundles of barrier fuel (see reference 5), both of standard nuclear design, will be used. Descriptions of the nuclear and mechanical designs of this fuel are contained in references 3, 4 and 5. Reference 3 also contains a complete set of references to topical reports which describe the GE analytical methods for nuclear, thermal-hydraulic transient and accident calculations and information regarding the applicability of these methods to cores containing a mixture of fuels. The use and safety implication of prepressurized fuel have been found acceptable in reference 4. The conclusions of reference 6 found that the methods of reference 3 were generally applicable to prepressurized fuel. Therefore, unless otherwise specified, reference 3, as supported by reference 6, is adequate justification for the current application of prepressurized fuel both for the barrier and nonbarrier fuel. Other aspects of the use of the barrier fuel demonstration bundles are also considered.

2.0 EVALUATION

We have reviewed the licensee's application and the associated proposed TS changes. The reload application follows the procedure described in reference 3, "Generic Reload Fuel Application." The thermal-hydraulic models and methodology used are those described in references 3 and 7.

2.1 Safety Limit MCPR; Thermal Hydraulics

The safety limit minimum critical power ratio (MCPR) is to assure at least 99.9% of the fuel rods in the core are not expected to experience boiling transition during anticipated operational transient events. As stated in reference 3, the safety limit MCPR (SLMCPR) is 1.07 for the core with retrofit 8x8 fuel, for both barrier and nonbarrier fuel. This limit has previously been found to be acceptable, as it is in this application.

2.2 Operating Limit MCPR; Use of ODYN Code

The most limiting operational transients for Cycle 6 for Quad Cities Unit 2 have been analyzed by the licensee to determine which event could potentially result in the largest reduction in the initial critical power ratio (ΔCPR). The ΔCPR values given in Section 11 of reference 2 are plant-specific values which include results for the transients calculated by using the ODYN methods (see references 7 and 8). The maximum value of ΔCPR resulting from the limiting transient, the generator load rejection without bypass transient, is 0.35 for Cycle 6 as compared to 0.23 for Cycle 5 (refs. 5 and 6). The large difference of ΔCPR for this transient is due to the use of the ODYN methods compared to the REDY methods used in Cycle 5.

The calculated ΔCPRs were adjusted to reflect either Option A or Option B ΔCPR by employing the conversion method described in references 7 and 8. The initial MCPRs are then determined by adding the ΔCPRs to the safety limit. Section 11 (reference 2) presents both the initial MCPRs for the nonpressurization events and adjusted initial MCPRs (Option A and Option B) for pressurization events. The maximum initial MCPRs (Option A and B) in Section 11 are specified as the operating limit MCPRs and are incorporated into the TSs. We have reviewed the operating limit MCPR results discussed above. These results are more limiting for Cycle 6 than for Cycle 5. We find these results acceptable.

The operating limit MCPR TS has been modified to include an Option B format where the operating limit MCPR varies with the measured scram time. The operating limiting MCPRs are incorporated in TSs 3.3.C/4.3.C and 3.5.K.

2.3 Thermal-Hydraulic Stability

The results of the thermal-hydraulic analysis (ref. 2) show that the maximum thermal-hydraulic stability decay ratio is 0.53 for Cycle 6 as compared to 0.52 for Cycle 5. Since operation in the natural circulation mode is prohibited by TS 2.1.A.4, there is additional margin to the core thermal-hydraulic stability, and we find the stability results acceptable for Cycle 6 operation.

2.4 ECCS Evaluation: MAPLHGR Limits

The previously approved reference document NEDO 24146A (see reference 12) contains an approved ECCS analysis for Quad Cities Unit 2, and continues to serve as the basis for generation of MAPLHGR limits for new fuel types. New MAPLHGR limits for the four barrier fuel types being loaded in the core for Cycle 6 are based on Addenda to reference 12 and were provided in the licensee's submittal (see reference 1). A non-barrier fuel type which is otherwise identical to one of the four barrier fuel types is also being loaded in the Cycle 6 core. The barrier fuel (of that type) MAPLHGR limits apply directly to the non-barrier fuel for the otherwise identical design.

MAPLHGR limits to non-prepressurized fuel have previously been conservatively applied to prepressurized fuel because of the unavailability of the slightly relaxed prepressurized MAPLHGR limits. The prepressured MAPLHGR limits are now available and are included for Cycle 6.

2.5 Pressure Safety Limit Changes Due to ATWS RPT

As of January 1, 1981, Quad Cities Unit 2 has had a recirculation pump trip (RPT) installed and implemented to mitigate the effects of an anticipated transient without scram (ATWS). While this modification reduces peak pressures for transients without scram, it also has the effect of increasing the peak pressurization for a severe transient with scram, such as load reject without bypass or a main steam isolation valve (MSIV) closure without valve position trip. However, pressurization transients which do cause the RPT setpoint (1250 psig) to be exceeded can cause higher steamdome pressures, where the measured vessel pressure limit is increased from 1325 psig to 1345 psig. The vessel peak pressure at the bottom of the vessel remains at 1375 psig. The assumed pressure difference of 30 psig still assures compliance with ASME code criteria of 110% of vessel design pressure (i.e. 110% x 1250 = 1375 psig).

Wording changes in the bases have also been incorporated to clarify that compliance of peak vessel pressure with the ASME criteria also assures compliance of the primary system piping with the USASI criteria for the limiting point (i.e. less than 1410 psig at the lowest point in the recirculation line). These changes were recommended by GE to remove the false implication in the current bases that all points in the primary system must remain less than the ASME criteria for the vessel (1375 psig) and are acceptable.

2.6 Barrier Fuel Demonstration

The planned demonstration irradiation of pellet/cladding interaction (PCI)-resistant BWR fuel involves a large scale (144 bundles) irradiation in Quad Cities Unit 2 starting with Cycle 6. It is proposed that about half (64) of the bundles would be power ramped, in groups of 16, i.e., one group of 16 would be ramped at the end of each of four successive reactor cycles.

The term "barrier fuel" stems from the use of a 0.003-inch thick, high purity zirconium liner, i.e., barrier which is metallurgically bonded to the Zircaloy-2 structural part of the fuel rod cladding. The dimensions of the fuel rods and the mechanical design of the fuel bundle are the same as the current GE prepressurized 8x8 retrofit bundle (P 8x8 R). A general description of the barrier fuel program including information on the program scope, fuel loading and operation, fuel mechanical design, and safety analyses was presented in a General Electric topical report, NEDO-24259 (ref. 5) which was reviewed and approved in October 1980.

In approving NEDO-24259 we stated (ref. 9) that the PCI barrier fuel demonstration was licensable, pending the receipt of further information to be submitted by the licensee in a reload analysis. That information would include (a) a detailed operating plan for the demonstration irradiation, (b) a commitment to perform on-line monitoring of fission product activity and post-irradiation examinations of the demonstration assemblies (consistent with GE recommendations), and (c) an estimate of the PCI failure probability (of the barrier fuel relative to standard fuel) that would coincide with each of the planned power ramps.

The licensee's responses to these conditional items are contained in references 10 and 11. These may be summarized as follows:

1. <u>Demonstration Irradiation Operating Plan and Analyses</u> - When more refined predictions are available (by June 1982), CECo will provide more detailed information on the expected peak local powers and power changes in the fuel that will be power ramped during the End of Cycle (EOC) 6 control rod withdrawal test. Those data shall indicate information on both the barrier fuel in the ramp cells (the cells for which the end of cycle power ramps are planned) as well as the adjacent fuel in the buffer regions.

2. <u>On-Line Monitoring and Post-Irradiation Examinations</u> - CECo will notify NRC Headquarters and the regional office should offgas activity increase during the EOC 6 ramp test to levels significantly in excess of the usual noise and transient behavior. In addition, provided that outage time is available off critical path, CECo will sip the test cell assemblies as well as any buffer region assemblies that are scheduled for reinsertion for Cycle 7 to confirm that the cladding is sound even if no failure indications were evident from offgas and coolant monitoring during the EOC 6 ramp test.

3. Estimate of Fuel Failure Probability - CECo will provide a comparison of the fuel failure probability for the planned ramp tests of the barrier fuel in the test cells relative to a postulated test with standard fuel in the test cells. That information will be supplied in June 1982 (at about mid-Cycle 6).

We have reviewed the licensee's responses and we believe that the information and commitments provided by CECo in references 10 and 11 are as detailed as possible at this time and that further definition can wait until mid-cycle when the actual EOC conditions and outage critical path are better known. We agree with CECo that the requested additional information on the items noted above, while related to the ramp tests, is not needed for review and approval of the reload licensing and Beginning of Cycle (BOC) 6 startup authorization. We, therefore, conclude that there is reasonable assurance that the proposed demonstration irradiation will not pose a threat to the public health and safety with regard to normal, steady-state operation of the barrier fuel and that the program is, therefore, acceptable.

3.0 Environmental Considerations

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 Section 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 the amendment.

4.0 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 this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: December 23, 1981

<u>References</u>

1. Letter, Schwartz (CECo) to Denton (NRC), July 27, 1981

2. Letter, Rausch (CECo) to Denton (NRC), August 21, 1981

3. "General Electric BWR Generic Reload Fuel Application," NEDE-24011-A-1, July 1979

4. Letter, Engel (GE) to Ippolito (NRC), January 30, 1979

5. "Barrier Fuel Demonstration Bundle Licensing," NEDO-24259A, May 1980

6. Letter, Ippolito (NRC) to Gridley (GE), April 16, 1979 and enclosed SER

7. Letter, Buckholz (GE) to Check (NRC), "Response to NRC Request for Information on ODYN Computer Model," September 5, 1980

8. Letter, Buckholz (GE) to Check (NRC), "ODYN Adjustment Methods for Determination of Operating Limits, January 19, 1981

9. Letter, Tedesco (NRC) to Engel (GE), November 12, 1980

10. Letter, Rausch (CECo) to Eisenhut (NRC), November 4, 1981

11. Letter, DelGeorge (CECo) to Eisenhut (NRC) December 3, 1981

12. "Loss-of-Coolant Accident Analysis Report for Dresden Units 2 & 3 and Quad Cities 1 & 2 Nuclear Power Station", Rev. 1, April 1979, and subsequent Errata and Addenda Nos. 1 thru 6.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-265

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. 69 to Facility Operating License DPR-30 issued to Commonwealth Edison Company and Iowa-Illinois Gas and Electric Company, which revised the Technical Specifications for operation of the Quad Cities Nuclear Power Station, Unit No. 2, located in Rock Island County, Illinois. The amendment becomes effective as of the date of issuance.

The amendment (1) authorizes operation in Cycle 6 using 224 assemblies of prepressurized 8 x 8R fuel, including 144 bundles of GE barrier fuel, (2) incorporates revised Minimum Critical Power Ratio (MCPR) limits in response to plant specific analyses for Cycle 6, (3) incorporates new Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits for the barrier fuel, (4) deletes MCPR, MAPLHGR and Linear Heat Generation Rate (LHGR) operating limits for all 7 x 7 fuel (none to remain in the core), and (5) changes the pressure safety limits due to the recently installed Anticipated Transients Without Scram Recirculation Pump Trip.

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 Section 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of the amendment.

For further details with respect to this action, see (1) the application for amendment dated July 27, 1981, as supplemented August 21, 1981, and December 3, 1981, (2) Amendment No. 69 to License No. DPR-30, 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, NW., Washington, D.C., and at the Moline Public Library, 504 - 17th Street, Moline, Illinois. 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 Licensing.

Dated at Bethesda, Maryland, this 23rd day of December 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

TA Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #2
Division of Licensing