June 17, 1988

Docket Nos.: 50-265

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Mr. Henry E. Bliss
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Bliss:

SUBJECT: CYCLE 10 RELOAD ANALYSIS AND LICENSE AMENDMENT (TAC 67605)

Re: Quad Cities Nuclear Power Station, Unit 2

The Commission has issued the enclosed Amendment No. 104 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station (QCNPS), Unit 2. This amendment is in response to your application dated March 28, 1988.

Based upon acceptable results of transient and accident analyses submitted for the Unit 2 reload, the Commission has approved revised Technical Specifications (TS) which change certain operating safety limits (i.e., LHGR, MCPR, MAPHLR, and RBM), expand the operating domain (ICF), and allow for continued operation with particular equipment out of service (i.e., SLO, FWHOOS, and RVOOS). In addition, license condition restrictions on coastdown operation and FFWTR have been deleted.

Although the core stability decay ratio for QCNPS indicates a stable core, CECo should be aware that the staff is still reviewing the LaSalle 2 instability event. All affected licensees will be informed of the final NRC position, which may include remedial action. It should also be noted, that the staff review of CECo's letter dated February 19, 1988 concerning presumptions made in the application of single failure criteria during this and previous ECCS analyses is not complete. Subsequent correspondence is forthcoming which will address Dresden and QCNPS conformance with NRC regulations of 10 CFR 50.46 and Appendix K with regard to the impact of a single passive or active DC bus failure upon the results of required ECCS analysis.

A copy of our Safety Evaluation is also enclosed. And a a Notice of Issuance will be included in the Commission's biweekly Federal Register.

Sincerely,

Leif J. Norrholm, Acting Director
Project Directorate III-2
Division of Reactor Projects - III, IV, V and Special Projects

Enclosures:
1. Amendment No. 104 to License No. DPR-30
2. Safety Evaluation

cc: w/enclosures:
See next page

PDIII-2:PM	PDIII-2:LA	PDIII-2	OGC
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Thierry M. Ross, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

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*See Previous Concurrence

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TRoss	LLuther	LNorholm	
6/ /88	6/ /88	6/17/88	6/ /88

Mr. Henry Bliss
Commonwealth Edison Company

Quad Cities Nuclear Power Station
Units 1 and 2

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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 NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 104
License No. DPR-30

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated March 28, 1988, 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). Furthermore, paragraph 3.C. of Facility Operating License No. DPR-29 is deleted in its entirety, and paragraph 3.B is hereby amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendix A and B, as revised through Amendment No.104, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. (Deleted)

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

FOR THE NUCLEAR REGULATORY COMMISSION

Leif J. Norrholm

Leif J. Norrholm, Acting Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 17, 1988

ATTACHMENT TO LICENSE AMENDMENT NO. 104

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by the captioned amendment number and contain marginal lines indicating the area of change.

REMOVE	INSERT
1.1/2.1-1	1.1/2.1-1
1.1/2.1-4	1.1/2.1-4
1.1/2.1-7	1.1/2.1-7
Figure 2.1-3	Figure 2.1-3
3.2/4.2-14	3.2/4.2-14
3.2/4.2-14a	3.2/4.2-14a
3.3/4.3-5	3.3/4.3-5
3.5/4.5-5	3.5/4.5-5
3.5/4.5-10	3.5/4.5-10
3.5/4.5-12	3.5/4.5-12
3.5/4.5-14	3.5/4.5-14
3.5/4.5-14a	3.5/4.5-14a
3.5/4.5-14b	3.5/4.5-14b
Figure 3.5-1 (Sheets 1 thru 6)	Figure 3.5-1 (Sheets 1 thru 5)
Figure 3.5-2	Figure 3.5-2
3.6/4.6-5a	3.6/4.6-5a
3.6/4.6-13a	3.6/4.6-13a

1.1/2.1 FUEL CLADDING INTEGRITY

SAFETY LIMIT

Applicability:

The safety limits established to preserve the fuel cladding integrity apply to those variables which monitor the fuel thermal behavior.

Objective:

The objective of the safety limits is to establish limits below which the integrity of the fuel cladding is preserved.

LIMITING SAFETY SYSTEM SETTING

Applicability:

The limiting safety system settings apply to trip settings of the instruments and devices which are provided to prevent the fuel cladding integrity safety limits from being exceeded.

Objective:

The objective of the limiting safety system settings is to define the level of the process variables at which automatic protective action is initiated to prevent the fuel cladding integrity safety limits from being exceeded.

SPECIFICATIONS

Safety Limit

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

The existence of a minimum critical power ratio (MCPR) less than 1.04 shall constitute violation of the fuel cladding integrity safety limit.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psig)

When the reactor pressure is ≤ 800 psig or core flow is less than 10% of rated, the core thermal power shall not exceed 25% of rated thermal power.

C. Power Transient

1. The neutron flux shall not exceed the scram setting established in Specification 2.1A for longer than 1.5 seconds as indicated by the process computer.

2. When the process computer is out of service, this safety limit shall be assumed to be exceeded if the neutron flux exceeds the scram setting established by Specification 2.1.A and a control rod scram does not occur.

Limiting Safety System Setting

A. Neutron Flux Trip Settings

The limiting safety system trip settings shall be as specified below:

1. APRM Flux Scram Trip Setting (Run Mode)

When the reactor mode switch is in the Run position, the APRM flux scram setting shall be as shown in Figure 2.1.1 and shall be:

$S \leq (.58W_D + 62)$

with a maximum setpoint of 120% for core flow equal to 98 x 10^6 lb/hr and greater.

where

S = setting in percent of rated power

W_D = percent of drive flow required to produce a rated core flow of 98 million lb/hr. In the event of operation with a maximum fraction of limiting power density (MFLPD) greater than the fraction of rated power (FRP), the setting shall be modified as follows:

$S \leq (.58W_D + 62 \left[\frac{FRP}{MFLPD} \right]$

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 for 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 setpoints 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 operation 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 temperature would increase to approximately 1100°F, which is below the perforation temperature of the cladding material. This had 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 of 13.4 kw/ft for fuel types P8x8R and BP8x8R, and 14.4 kw/ft for fuel types GE8x8E and GE8x8EB. 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.

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 in accordance with Regulatory Guide 1.49. 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.

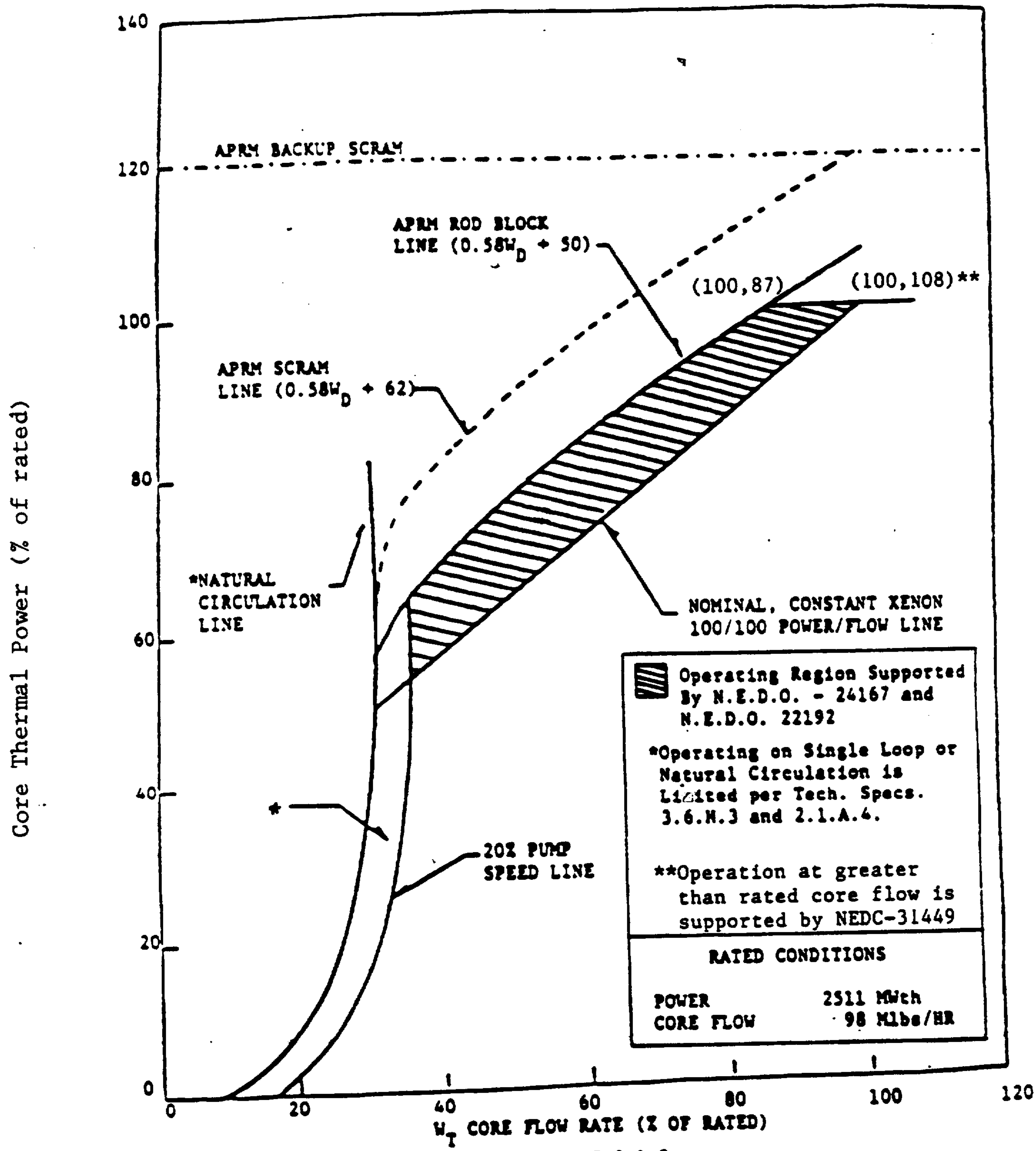


FIGURE 2.1-3
(SCHEMATIC)
APRM FLOW BIAS SCRAM RELATIONSHIP
TO NORMAL OPERATING CONDITIONS

TABLE 3.2-3

INSTRUMENTATION THAT INITIATES ROD BLOCK

Minimum Number of Operable or Tripped Instrument Channels per Trip System (1)	Instrument	Trip Level Setting
2	APRM upscale (flow bias)[7]	$\leq[0.58W_D + 50]\ \frac{FRP}{MFLPD}$ [2]
2	APRM upscale (Refuel and Startup/Hot Standby mode)	≤12/125 full scale
2	APRM downscale[7]	≥3/125 full scale
1	Rod block monitor upscale (flow bias)[7]	$\leq 0.65W_D + 43$[2][10]
1	Rod block monitor downscale[7]	≥3/125 full scale
3	IRM downscale[3] [8]	≥3/125 full scale
3	IRM upscale[8]	≤108/125 full scale
2[5]	SRM detector not in Startup position [4]	≥2 feet below core centerline
3	IRM detector not in Startup position [8]	≥2 feet below core centerline
2[5] [6]	SRM upscale	$\leq 10^5$ counts/sec
2[5]	SRM downscale [9]	$\geq 10^2$ counts/sec
1 (per bank)	High water level in scram discharge volume (SDV)	≤ 25 gallons (per bank)
1	SDV high water level scram trip bypassed	NA

Notes

1. For the Startup/Hot Standby and Run positions of the reactor mode selector switch, there shall be two operable or tripped trip systems for each function except the SRM rod blocks. IRM upscale and IRM downscale need not be operable in the Run position, APRM downscale, APRM upscale (flow biased), and RBM downscale need not be operable in the Startup/Hot Standby mode. The RBM upscale need not be operable at less than 30% rated thermal power. One channel may be bypassed above 30% rated thermal power provided that a limiting control rod pattern does not exist. For systems with more than one channel per trip system, if the first column cannot be met for one of the two trip systems, this condition may exist for up to 7 days provided that during that time the operable system is functionally tested immediately and daily thereafter; if this condition lasts longer than 7 days the system shall be tripped. If the first column cannot be met for both trip systems, the systems shall be tripped.

2. W_D is the percent of drive flow required to produce a rated core flow of 98 million lb/hr. Trip level setting is in percent of rated power (2511 MWt).

3. IRM downscale may be bypassed when it is on its lowest range.

4. This function is bypassed when the count rate is $\geq$ 100 cps.

5. One of the four SRM inputs may be bypassed.

6. This SRM function may be bypassed in the high IRM ranges (ranges 8, 9, and 10) when the IRM upscale rod block is operable.

7. Not required to be operable when performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 MWt.

8. This IRM function occurs when the reactor mode switch is in the Refuel or Startup/Hot Standby position.

9. This trip is bypassed when the SRM is fully inserted.

10. The trip level setting shall be a maximum of 108% for core flow equal to 98 x 10^6 lb/hr and greater.

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 0.68 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 that 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% Δ k. 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

Prior to entering EGC and once per shift while operating in EGC, the EGC operating parameters will be reviewed for acceptability.

provided that during such 7 days all active components of the automatic pressure relief subsystems, the core spray subsystems, LPCI mode of the RHR system, and the RCIC system are operable.

3. If the requirements of Specification 3.5.C cannot be met, an orderly shutdown shall be initiated, and the reactor pressure shall be reduced to 90 psig within 24 hours.

D. Automatic Pressure Relief Subsystems

1. The automatic pressure relief subsystem shall be operable whenever the reactor pressure is greater than 90 psig, irradiated fuel is in the reactor vessel and prior to reactor startup from a cold condition.

2. From and after the date that two of the five relief valves of the automatic pressure relief subsystem are made or found to be inoperable when the reactor is pressurized above 90 psig with irradiated fuel in the reactor vessel, reactor operation is permissible only during the succeeding 7 days unless repairs are made and provided that during such time the HPCI subsystem is operable.

3. If the requirements of Specification 3.5.D cannot be met, an orderly shutdown shall be initiated and the reactor pressure shall be reduced to 90 psig within 24 hours.

operable immediately. The RCIC system shall be demonstrated to be operable daily thereafter. Daily demonstration of the automatic pressure relief subsystem operability is not required provided that two feedwater pumps are operating at levels above 300 MWe; and one feedwater pump is operating as normally required with one additional feedwater pump operable at power levels less than 300 MWe.

D. Automatic Pressure Relief Subsystems.

Surveillance of the automatic pressure relief subsystem shall be performed as follows:

1. The following surveillance shall be carried out on a six-month surveillance interval:

   a. With the reactor at pressure each relief valve shall be manually opened. Relief valve opening shall be verified by a compensating turbine bypass valve or control valve closure.

2. A logic system functional test shall be performed each refueling outage.

3. A simulated automatic initiation which opens all pilot valves shall be performed each refueling outage.

4. When it is determined that two relief valves of the automatic pressure relief subsystem are inoperable, the HPCI shall be demonstrated to be operable immediately.

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 is 13.4 kw/ft. for fuel types P8x8R and BP8x8R. For fuel types GE8x8E and GE8x8EB the maximum allowable LHGR is 14.4 kw/ft.

K. Minimum Critical Power Ratio (MCPR)

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

1.30 for $\tau_{AVE} \leq 0.68$ sec

1.35 for $\tau_{AVE} \geq 0.86$ sec

$0.278\ \tau_{AVE} + 1.111$

for $0.68 \text{ sec} \leq \tau_{AVE} \leq .86$ sec

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.

Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, which is tested daily, a 7-day repair period was specified.

C. High-Pressure Coolant Injection

The high-pressure coolant injection subsystem is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI mode of the RHR system or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of offsite electrical power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled, thus no cladding damage occurs (reference SAR Section 6.2.5.3). The repair times for the limiting conditions of operation were set considering the use of the HPCI as part of the isolation cooling system.

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystems are a backup to the HPCI subsystem. They enable the core spray subsystem and LPCI mode of the RHR system to provide protection against the small pipe break in the event of HPCI failure by depressurizing the reactor vessel rapidly enough to actuate the core spray subsystems and LPCI mode of the RHR system. The core spray subsystem and/or the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures to less than 2200°F, to assure that core geometry remains intact, to limit the core wide clad metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

Analyses have shown that only four of the five valves in the automatic depressurization system are required to operate. Loss of one of the relief valves does not significantly affect the pressure relieving capability, therefore continued operation is acceptable. Loss of two relief valves significantly reduces the pressure relief capability of the ADS: thus, a 7 day repair period is specified with the HPCI available, and a 24 hour repair period with the HPCI unavailable.

E. RCIC

The RCIC system is provided to supply continuous makeup water to the reactor core when the reactor is isolated from the turbine and when the feedwater system is not available. Under these conditions the pumping capacity of the RCIC system is sufficient to maintain the water level above the core without any other water system in operation. If the water level in the reactor vessel decreases to the RCIC initiation level, the system automatically starts. The system may also be manually initiated at any time.

The HPCI system provides an alternate method of supplying makeup water to the reactor should the normal feedwater become unavailable. Therefore, the specification calls for an operability check of the HPCI system should the RCIC system be found to be inoperable.

F. Emergency Cooling Availability

The purpose of Specification 3.5.F is to assure a minimum of core cooling equipment is available at all times. If, for example, one core spray were out of service and the diesel which powered the opposite core spray were out of service, only two RHR pumps would be available. Likewise, if two RHR pumps were out of service and two RHR service water pumps on the opposite side were also out of service no containment cooling would be available. It is during the refueling outages that major maintenance is performed and during such time that all low-pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation systems. Thus, the specification precludes the events which could require core cooling. Specification 3.9 must also be consulted to determine other requirements for the diesel generators.

Quad-Cities Units 1 and 2 share certain process systems such as the makeup demineralizers and the radwaste system and also some safety systems such as the standby gas treatment system, batteries, and 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 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 plus two percent for uncertainty 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 and 4. 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 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 0.68 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. "Quad Cities Nuclear Power Station Units 1 & 2 SAFER/GESTR-LOCA Loss of Coolant Accident Analysis" NEDC-31345P.*

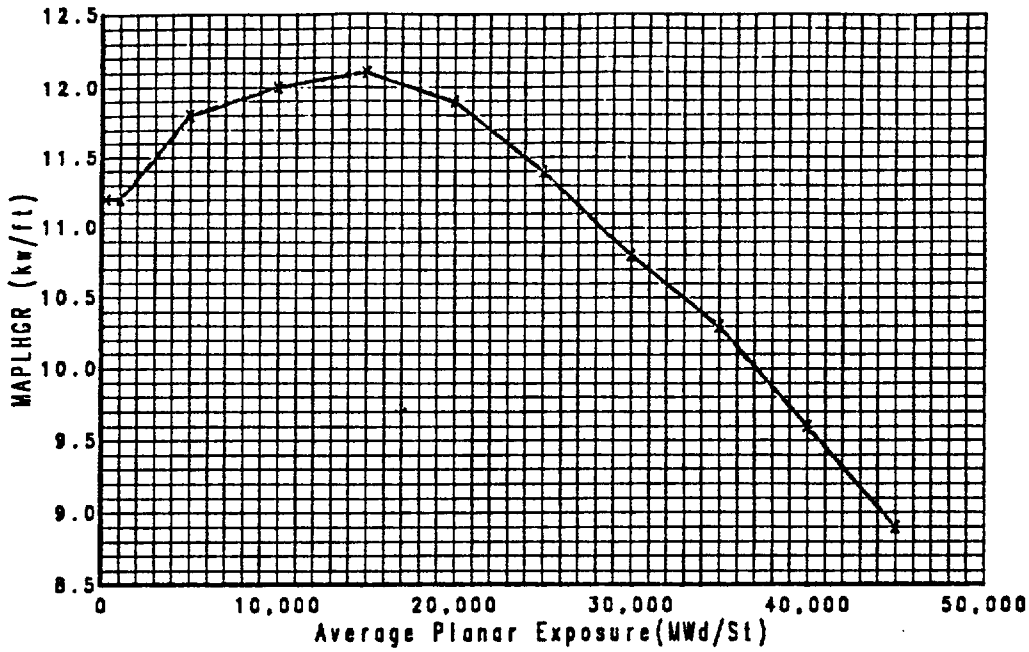
2. "Generic Reload Fuel Application," NEDE-24011-P-A**

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).

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

MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB282



MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB283H

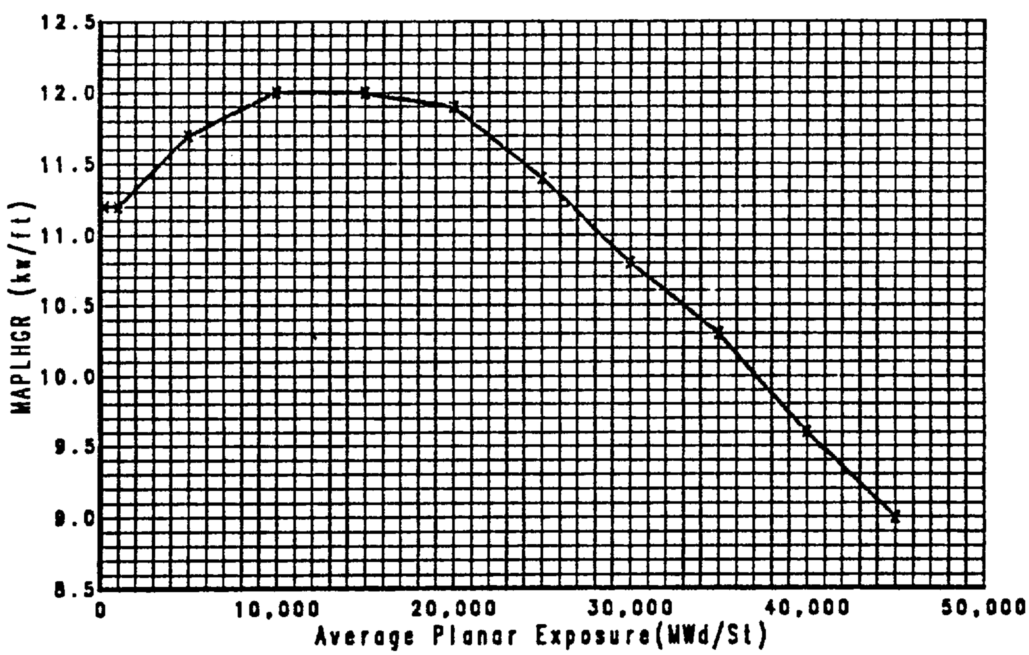
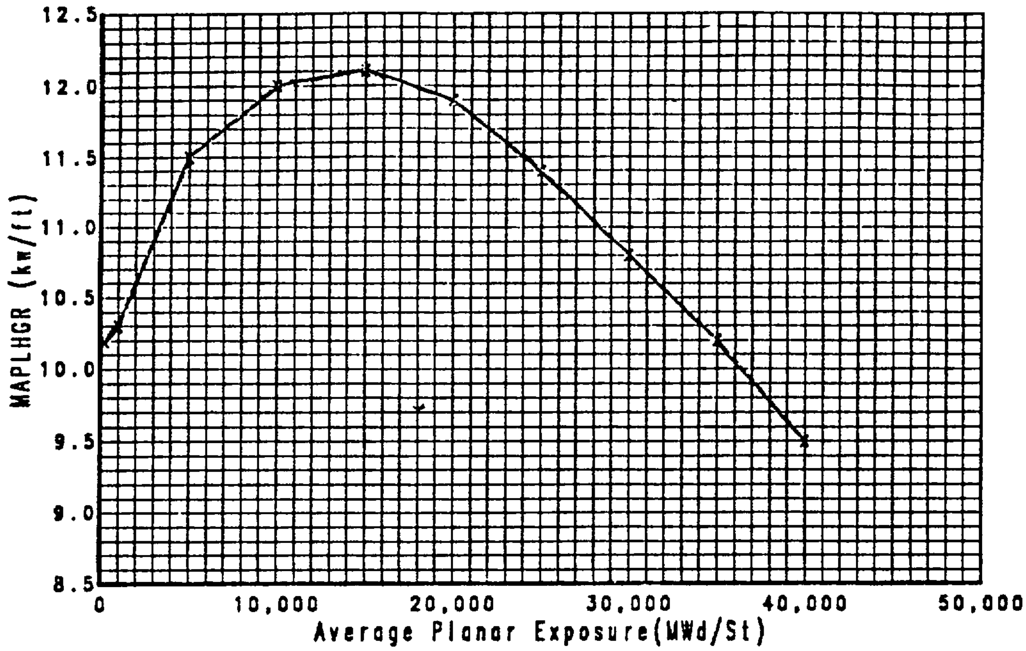


Figure 3.5-1
Sheet 1 of 5

MAPLHGR VS. Average Planar Exposure
Fuel Type P8DGB263L



MAPLHGR VS. Average Planar Exposure
Fuel Type P8DGB298

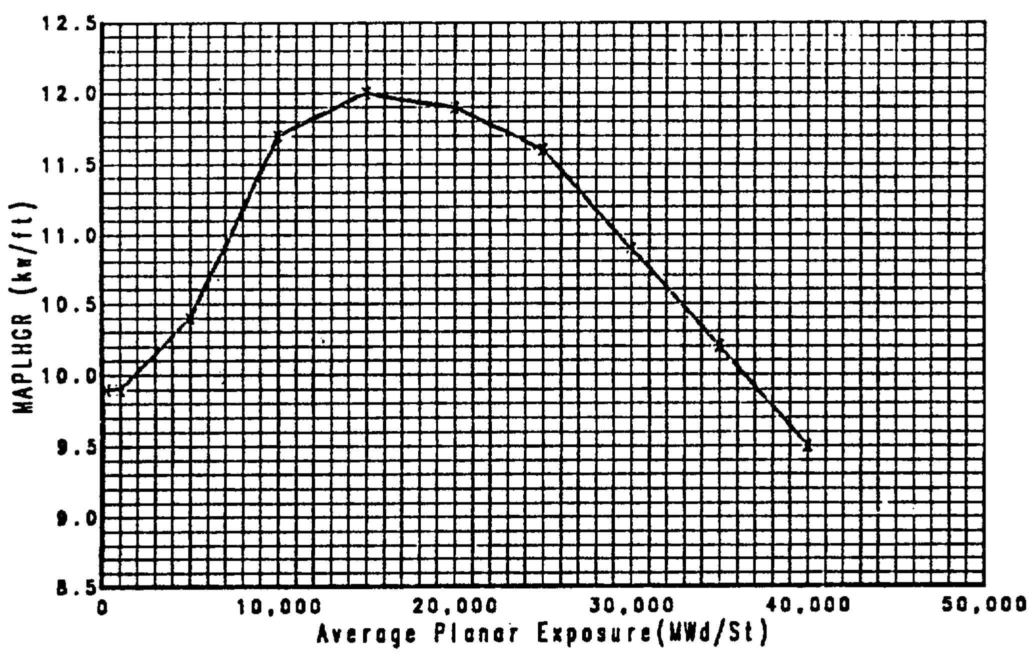
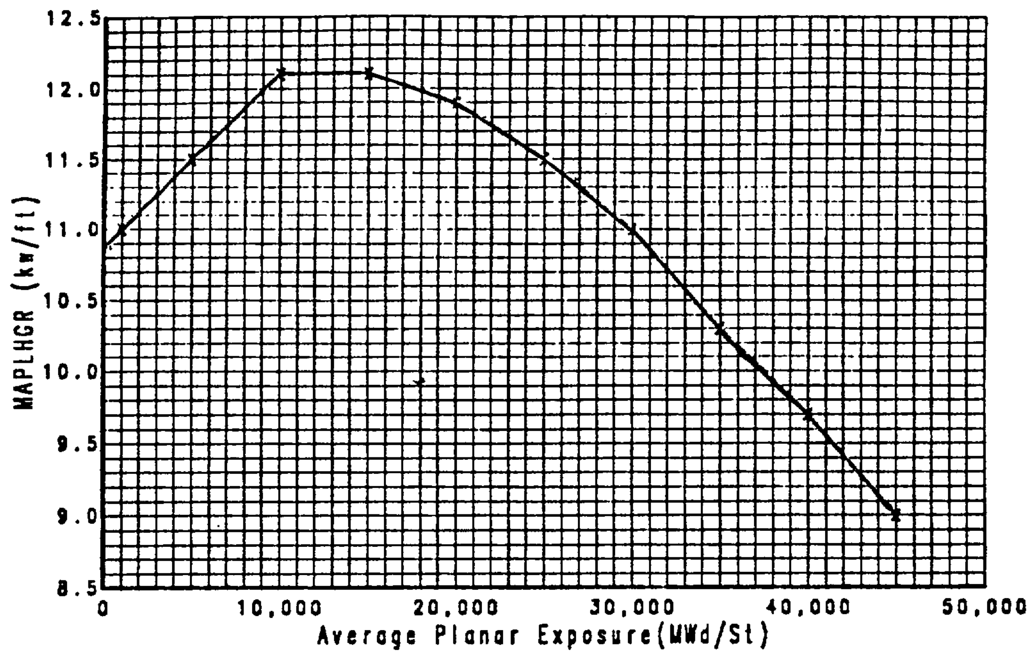


Figure 3.5-1
Sheet 2 of 5

MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB299



MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB299L

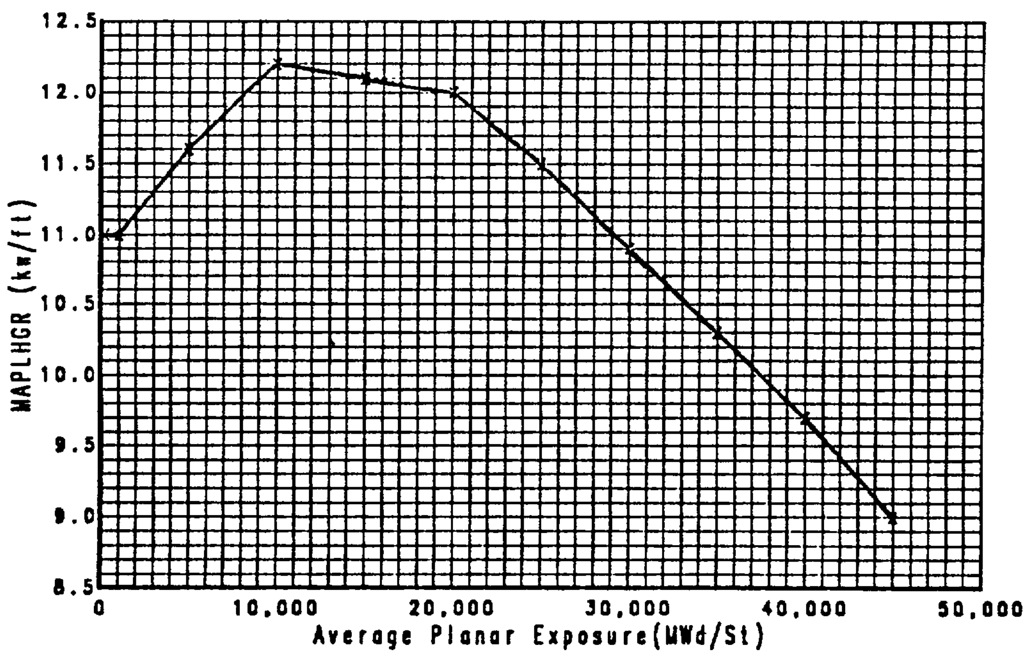
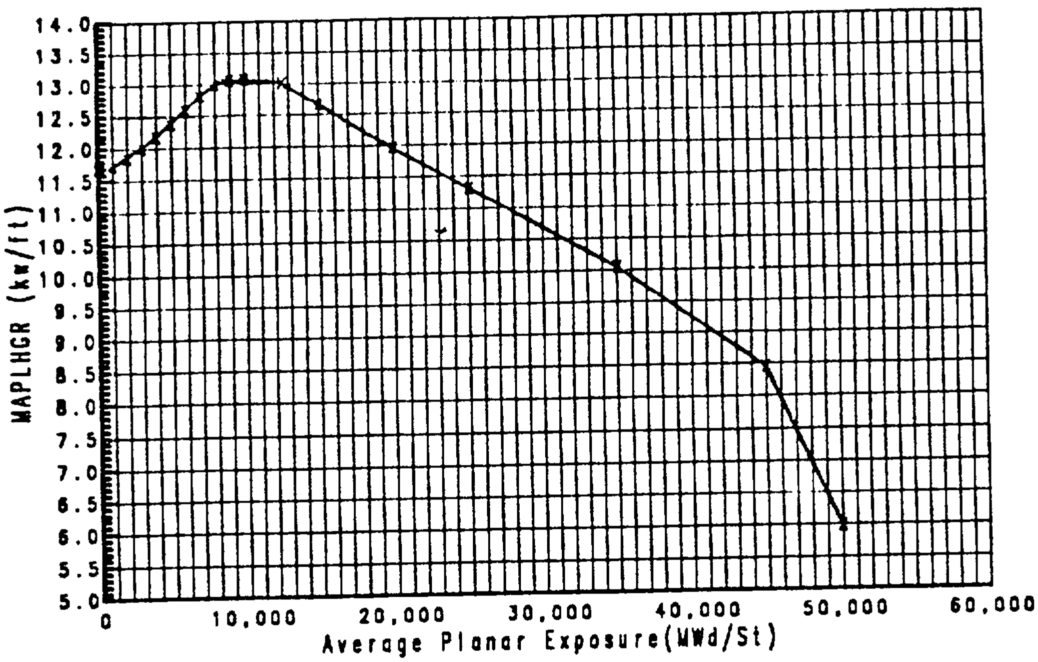


Figure 3.5-1
Sheet 3 of 5

MAPLHGR VS. Average Planar Exposure
Fuel Type BD316A



MAPLHGR VS. Average Planar Exposure
Fuel Type BD300C

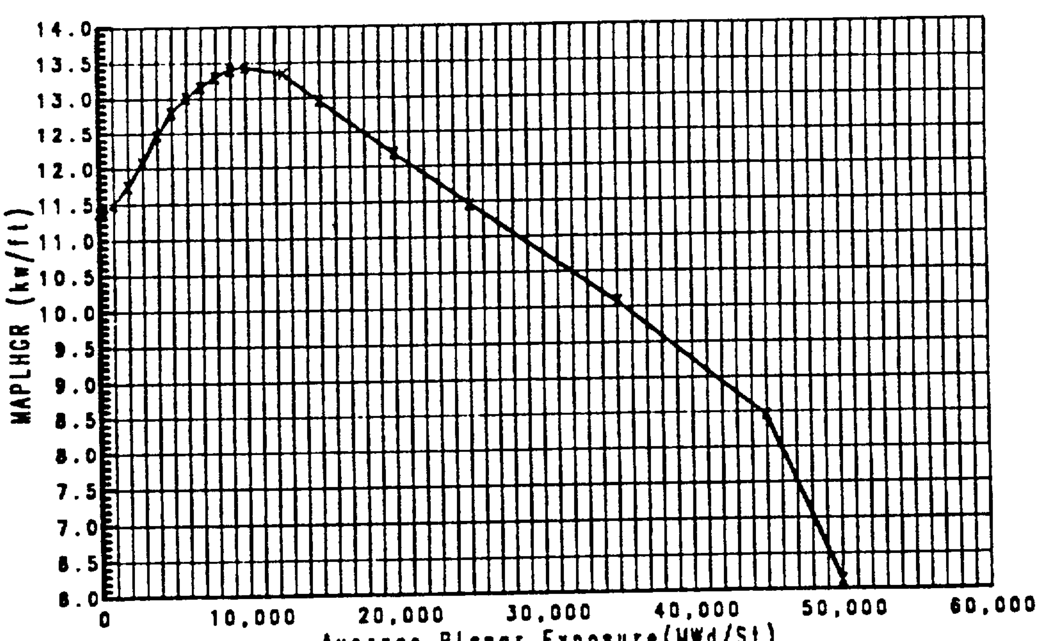


Figure 3.5-1
Sheet 4 of 5

MAPLHGR VS. Average Planar Exposure
Fuel Types P8DRB265H, BP8DRB265H

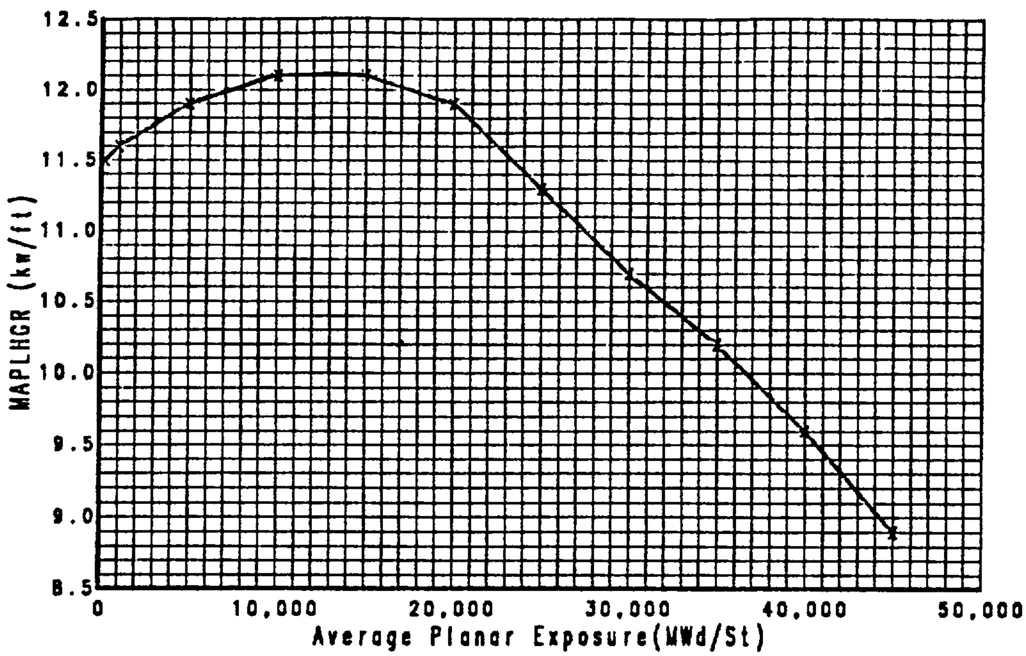


Figure 3.5-1
Sheet 5 of 5

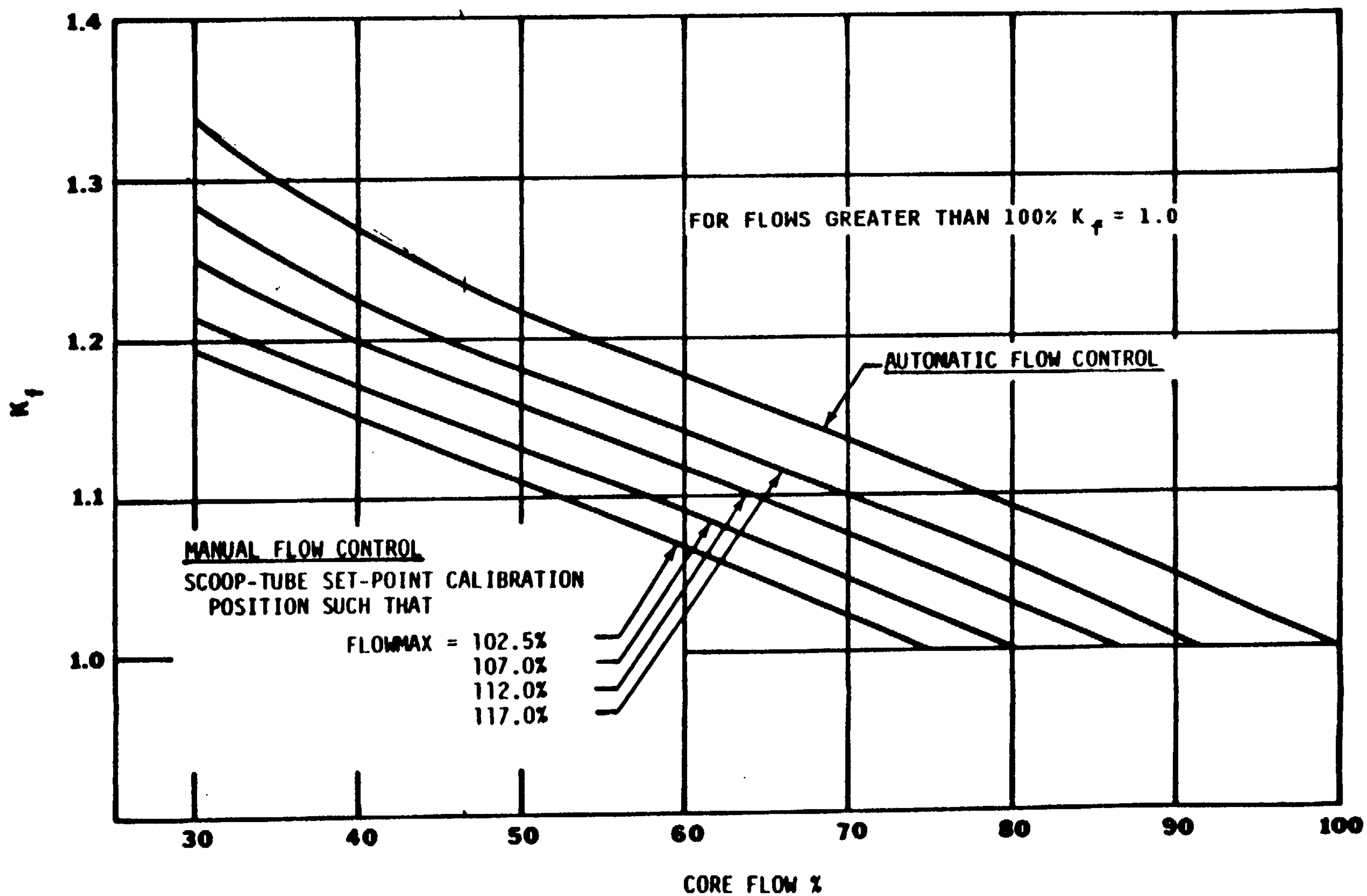


FIGURE 3.5-2
K_f FACTOR

3. Prior to Single Loop Operation for more than 12 hours, the following restrictions are required:

   a. The MCPR Safety Limit shall be increased by 0.01. (T.S. 1.1A);

   b. The MCPR Operating Limit shall be increased by 0.01 (T.S. 3.5.K);

   c. The flow biased APRM Scram and Rod Block Setpoints shall be reduced by 3.5% to read as follows:

      T.S. 2.1.A.1;
      S $\leq$ .58WD + 58.5

      T.S. 2.1.A.1;*
      S $\leq$ (.58WD + 58.5) FRP/MFLPD

      T.S. 2.1.B;
      S $\leq$ .58WD + 46.5

      T.S. 2.1.B;*
      S $\leq$ (.58WD + 46.5) FRP/MFLPD

      T.S. 3.2.C (Table 3.2-3);*
      APRM upscale $\leq$ (.58WD + 46.5) FRP/MFLPD

* In the event that MFLPD exceeds FRP.

   d. The flow biased RBM Rod Block setpoints shall be reduced by 4.0% to read as follows:

      T.S. 3.2.C (Table 3.2-3); RBM Upscale $\leq$ .65WD + 39

   e. The suction valve in the idle loop shall be closed and electrically isolated except when the idle loop is being prepared for return to service.

The licensee's analyses indicate that above 80% power the loop select logic could not be expected to function at a speed differential of 15%. Below 80% power, the loop select logic would not be expected to function at a speed differential of 20%. This specification provides a margin of 5% in pump speed differential before a problem could arise. If the reactor is operating on one pump, the loop select logic trips that pump before making the loop selection.

Analyses have been performed which support indefinite single loop operation provided the appropriate restrictions are implemented within 12 hours. The MCPR Safety Limit has been increased by 0.01 to account for core flow and TIP reading uncertainties which are used in the statistical analysis of the safety limit. The MCPR Operating Limit has also been increased by 0.01 to maintain the same margin to the safety limit as during Dual Loop operation.

The flow biased scram and rod block setpoints are reduced to account for uncertainties associated with backflow through the idle jet pumps when the operating recirculation pump is above 20 - 40% of rated speed. This assures that the flow biased trips and blocks occur at conservative neutron flux levels for a given core flow.

The closure of the suction valve in the idle loop prevents the loss of LPCI flow through the idle recirculation pump into the downcomer.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 104 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

DOCKET NO. 265

1.0 INTRODUCTION

By letter from J. A. Silady, Commonwealth Edison Company (CECo), to USNRC, dated March 28, 1988 (Ref. 1), Technical Specification changes were proposed for the operation of Quad Cities Station Unit 2 for Cycle 10 (QC2C10) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested Technical Specification (TS) changes and reports (including Reference 2 through 4) discussing the reload and analysis done to support and justify Cycle 10 operation including an increased flow region, equipment out of service and single loop operation.

The reload for Cycle 10 is generally a normal reload with no unusual core features or characteristics. Proposed TS changes relate to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits for the new fuel, MAPLHGR and Minimum Critical Power Ratio (MCPR) limits for all of the fuel using Cycle 10 core and transient parameters, extended operating regions and conditions, and new approved analytical methods. The new fuel is the extended burnup type which has been approved for use in several recent GE reloads.

The submittal proposes an extension of the current allowable operating region on the reactor power-flow map via an increased core flow (ICF) extension. Extended Load Line Limit Analysis (ELLA) and associated TS have also been proposed for Quad Cities Unit 2.

Also proposed for the cycle and supported with GE analyses is operation with "equipment-out-of-service" extended operating modes including feedwater heaters out of service (FWHOOS), final feedwater temperature reduction (FFWTR), relief valve out of service (RVOOS) and single loop operation (SLO). TS MCPR limits bounding analyzed combinations of these conditions have been proposed.

2.0 EVALUATION

2.1 Reload Description

The QC2C10 will retain 8 P8DGB263L and 24 P8DGB298 GE fuel assemblies from Cycle 6, 200 BP8DRB265H GE fuel assemblies from Cycle 7, 72 BP8DRB282 and 104 BP8DRB283H GE fuel assemblies from Cycle 8, 64 BP8DRB299 and 68 BP8DRB299L GE fuel assemblies from a previous cycle, and add 92 BD300C and 72 BD316A new GE8x8EB fuel assemblies. The reload is based on a previous end of cycle core nominal average exposure of 21,666 MWd/MT and Cycle 10 end of cycle exposure of 22,754 MWd/MT. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery.

2.2 Fuel Design

The new fuel for Cycle 10 is the GE extended burnup fuel GE8x8EB. The fuel designations are BD300C and BD316A. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Refs. 5 & 6). The specific description of this fuel has been accepted and the fuel description is also presented for QC2C10 in Reference 4. This fuel description is acceptable.

LOCA analyses have been done for the retained and reload fuel using the improved SAFER/GESTR-LOCA methods approved by the staff. The initial condition MAPLHGR values used in these analyses are less restrictive than those used in the fuel mechanical integrity design analyses. Thus the multi-axial region MAPLHGR TS used in some other recent reload applications of extended burnup fuel are unnecessary, and only a single set of burnup dependent values, for each fuel type, as determined by the mechanical design are required. The MAPLHGR values for both the reload and retained fuel have been calculated with approved methodology (GESTAR II, Reference 6, Section 2 of Vol. 1) and are acceptable.

The proposed LHGR limit for the GE8x8EB fuel is 14.4 KW/ft (rather than the 13.4 for other GE fuel). The LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Ref. 5). This LHGR is acceptable for the GE fuel in QC2C10.

2.3 Nuclear Design

The nuclear design for QC2C10 has been performed by GE with the approved methodology described in GESTAR II (Ref. 6). The results of these analyses are given in the GE reload report (Ref. 2) in standard GESTAR II format. The results are within the range of those usually encountered for BWR reloads. In particular, the shutdown margin is 4.2% delta-k at the beginning of cycle and 1.2% delta-k at the minimum conditions, thus fully meeting the required 0.38% delta-k shutdown margin. The standby liquid control system also meets shutdown requirement with a reasonable shutdown margin of 4.3% delta-k. Since these and other QC2C10 nuclear design parameters have been obtained with previously approved methods, and fall within expected ranges, the nuclear design is acceptable.

2.4 Thermal-Hydraulic Design

The thermal-hydraulic design for QC2C10 has been performed by GE with the approved methodology described in GESTAR II (Ref. 6) and the results are given in the GE reload report (Ref. 2). The GEMINI/ODYN transient analysis methodology (Ref. 6) was used for relevant transient analysis.

The Operating Limit MCPR (OLMCPR) values are determined by the limiting transients, which, for standard conditions, are usually Rod Withdrawal Error (RWE), Feedwater Controller Failure (FWCF), Turbine Trip Without Bypass (TTNBP) and Load Rejection Without Bypass (LRNBP). The analyses of these events for QC2C10, using the standard approved GEMINI/ODYN Option A and B approach for pressurization transients in standard and extended operating regions and with analyzed equipment out of service combinations, provide new Cycle 10 TS values of OLMCPR as a function of average scram time. For all standard operating conditions TTNBP is controlling at both Option A and B limits, giving OLMCPR values of 1.31 and 1.27, respectively. However, to accommodate the extended and equipment out-of-service conditions the OLMCPR has been analyzed (Ref. 3) for those conditions also. This has resulted in an increase to 1.35 for Option A and 1.30 for Option B associated with the feedwater heater out-of-service (FWHOOS) analyses. Approved methods (Ref. 6) were used to analyze these events; analyses and results are acceptable, and fall within expected ranges.

GE has calculated the core stability decay ratio at the point of minimum stability (the intersection of the natural circulation line and the extended APRM block line) for QC2C10. The calculated value of reactor core stability decay ratio is 0.58. This indicates a stable core since there is substantial margin to the acceptable value of 0.8 (for approved GE methods). However, due to the LaSalle 2 instability event which demonstrated that the decay ratio acceptance criteria do not provide assurance of core stability, the licensee will be informed of any remedial action to be taken upon the completion of our review of generic implications of the LaSalle event.

2.5 Transient and Accident Analysis

The transient and accident analysis methodologies used for QC2C10 are described and NRC approval indicated in GESTAR II (Ref. 6). The GEMINI/ODYN method was used for the core wide transient analysis which includes load rejection without bypass (LRNBP), loss of feedwater heating and feedwater controller failure. The local rod withdrawal error (RWE) was analyzed on a plant and cycle specific basis and a rod block setpoint of 108% was selected to provide an OLMCPR of 1.24 for all fuel types. This is less than the core wide events. The limiting MCPR events for QC2C10 are indicated in Section 2.4. The core wide and local transient analysis methodologies and results are acceptable and fall within expected ranges.

The limiting pressurization event, the main steam isolation valve closure with flux scram, analyzed with standard GESTAR II methods, gave results for peak steam dome and vessel pressures for standard and extended operating regions and equipment out-of-service conditions well under required limits. These are acceptable methodologies and results.

Banked position withdrawal sequence and rod patterns are used for Quad Cities 2. For plants using this system the Rod Drop Accident (RDA) event has been statistically analyzed generically and it was found that with a high degree of confidence the peak fuel enthalpy would not approach the NRC limit of 280 cal/gm for this event. This approach and analysis has been approved by NRC (Ref. 6). This approach is acceptable for QC2C10.

The LOCA analyses for QC2C10 were performed using the SAFER/GESTR-LOCA methodology. This methodology (Refs. 6 & 7) has been approved by the staff and used and approved in several recent reload applications. The licensee has reported the results of these analyses (Ref. 4) which are required to meet the necessary conditions (Ref. 7). Specifically, the analyses include break sizes from 0.05 ft^2 to the maximum DBA recirculation suction line break (4.26 ft^2). Seven different break sizes were analyzed (for either nominal input or Appendix K values) in conjunction with ECCS failure combinations. A total of 14 cases were evaluated to establish the trend of PCT curves (nominal and Appendix K) versus break size.

The input parameters for both the nominal and Appendix K cases are within those used in the approved generic analyses. The ECCS configuration of Quad Cities 2 (4 Low Pressure Coolant Injection, 2 Low Pressure Core Spray, High Pressure Coolant Injection, Automatic Depressurization System) is consistent with the ECCS configuration of a generic BWR-3/4. The results show that the DBA recirculation suction line break with battery failure is the limiting case. The plant-specific Appendix K calculation demonstrate that the DG/HPCI failure is limiting for the P8x8R fuel, which is the limiting fuel type. The calculated PCT is 828°F when nominal input values are used and 1382°F when Appendix K input values (plus adder) are used. Because the accident analyses have been performed using approved methods, and the results meet the staff's acceptance criteria, we conclude that these analyses are acceptable.

LOCA sensitivity studies or specific calculations were examined to consider the effect of extended or equipment out-of-service operation (Refs. 3 & 4). This included the full range discussed in Section 2.6. The changes to peak cladding temperature were generally small (or the condition was included in the base calculations, e.g., RVOOS) compared to the large margins available, so that no modifications to MAPLHGR limits are required for these conditions. These results are reasonable and acceptable. The results indicate that the TS MAPLHGR limits are not set by the LOCA calculations but by the thermal-mechanical design calculations.

2.6 Operating Extensions and Equipment out-of-Service

The QC2C10 reload submittal proposes extensions to standard operating regions and equipment out-of-service in the GESTAR II standard category of "Operating Flexibility or Margin Improvement Options." The selected options are ICF, FFWTR, FWHOOS, RVOOS and SLO. These have become commonly selected and approved options for a number of reactors in recent years. These options and associated analyses, including relevant transients and accidents, are described and discussed in Reference 3. Included in the analysis and discussion is the application for operation beyond nominal end of cycle with ICF (or decreased flow) and FFWTR, and coastdown to lower power levels (as low as 20 percent is assumed). The coastdown power and feedwater temperature reduction and the SLO analysis are intended to provide a basis for the removal of Quad Cities 2 license restrictions and for SLO TS additions.

For ICF the analyses are performed at the bounding condition of 108% of rated core flow (Ref. 3). The proposed operating region is bounded by the 108% APRM rod block line (0.58 W + 50%), the rated power line and the rated rod line. The region of operation above the rated rod line is known as the Extended Load Line Limit Analysis (ELLLA) region. The Safety Evaluation for this operating region includes operation beyond normal end-of-cycle, up to 100°F FFWTR (with ICF or reduced flow) and power coastdown (20 percent assumed in the analysis). Conservative power profiles were assumed. The transient analyses were used to determine OLMCPR values for these operating conditions. As discussed in Section 2.4, OLMCPR for QC2C10 is determined by the analysis of FWHOOS. The LOCA examination concluded that the effects on MAPLHGR were insignificant compared to the large margin available. The core stability is addressed in Section 2.4. The effects of ICF and FFWTR related loads, vibration, and fatigue on various reactor internals, and the impact on containment LOCA response, was examined and were found to be within allowable design limits except for (as is usually the case) a possible need for a slightly reduced feedwater nozzle refurbishment interval (based on seal leakage). Throughout these analyses the transients and accident examined, the methodologies and the results were completely similar to those reviewed on previous approved ICF-FFWTR applications for other reactors. The analyses and results and operation in this extended region are acceptable for Quad Cities 2.

The FWHOOS was analyzed in a similar manner. It is similar to FFWTR except for potential duration and time of occurrence in cycle which can affect core parameters to a greater extent. As indicated in Section 2.4, the extreme conditions used for analysis resulted in setting the OLMCPR for QC2C10. The increased limit is caused primarily by changes in axial power distribution and resulting effectiveness of scram action. This review concludes that operation with FWHOOS is acceptable for QC2C10.

For RVOOS the limiting pressurization event was analyzed and evaluated with the lowest setpoint safety relief valve OOS. The impact on MCPR is negligible. Standard sensitivity studies also show the effect on overpressure is small and results in adequate margin. The effect of a relief valve out of service was included in the LOCA analyses. It is concluded that operation with one RVOOS is acceptable.

Single loop operation (SLO) analysis was previously reviewed and approved by USNRC. Previous SLO analysis demonstrated that, within the normal operating domain and without equipment out-of-service, the consequences of abnormal operation transients from one-loop operation will be considerably less severe than those analyzed for a two-loop operation mode. MAPLHGR changes for QC2 are not necessary here since, as previously indicated (Section 2.4), the LOCA analysis for SLO (using the new methodology) provides peak cladding temperature well below limits. The stability issue for QC2 core (GE8x8EB fuel) should follow the staff position stated in Section 2.4 of this SER.

2.7 Technical Specification

The following TS changes have been proposed for Quad Cities 2 to implement the reload analyses and operation changes which have been discussed. The reason or bases for the changes have been for the most part already discussed and approved and the changes will only be briefly described as follows:

1. License Restriction 3.C

   Remove restrictions on coastdown operation and off-normal feedwater heating. These including coastdown to 20% and coastdown with off-normal FW heating have been analyzed by GE using approved methods to determine the operating restrictions (MCPR, MAPLHGR) which are bounded by the previous cycle. Therefore, the proposed change is acceptable.

2. TS 1.1.A on Page 1.1/2.1-1

   Reduction of the MCPR fuel cladding safety limit from 1.07 to 1.04 as generically approved by the NRC for the GE8x8EB fuel. This is acceptable since Quad Cities 2 is a D-lattice plant with Cycle 10 being the third successive reload core with high bundle R-factor ($\geqq$ 1.04) fuel design (based on an improved analysis described in the approved Amendment 14 to NEDE-24011-PA).

3. Basis 1.1.A on page 1.1/2.1-4 and TS 3.5.J. on page 3.5/4.5-10

   Delete 7x7 discussion since it is no longer in use for this Cycle 10 reload and include the new LHGR limit of 14.4 KW/ft for new addition of the GE8x8EB fuel types. This is acceptable.

4. Basis 2.1 on page 1.1/2.1-7

Change analyzed conditions from "up to the rated thermal power condition of 2511 MWt" to "in accordance with Regulatory Guide 1.49" which states that transients must be analyzed up to 102% rated core thermal power. This is acceptable.

5. Figure 2.1-3

Add operating region as defined by the increased core flow analysis which is evaluated in Section 2.6 of this SER. This is acceptable.

6. Table 3.2-3

The Technical Specification for RBM upscale trip level setting change from 0.65 Wd + 42 to 0.65 Wd + 43 so that at 100% drive flow the rod block setting is equal to 108% core flow which is equal to 98×10^6 lb/hr. This is acceptable.

7. TS 3.3.C.5 on page 3.3/4.3-5, TS 3.5.K on Page 3.5/4.5-10, and TS Bases on page 3.5/4.5-14a

The 20 percent scram insertion time is changed to 0.68 seconds corresponding to the ODYN B analysis. MCPR limits are revised in accordance with analysis results from approved GEMINI/ODYN methodology.

8. TS 3.5.D.2 and 4.5.D.4 on page 3.5/4.5-5, and TS Bases on page 3.5/4.5-12

The limiting conditions for operation and surveillance requirements and for operation basis for the automatic pressure relief subsystem are changed to reflect the analysis for continued operation with one relief valve out of service (RVOOS). It allows extended operation with one RVOOS and limited operation (7 days) with two RVOOS provided HPCI is demonstrated to be operable. Also, change the word "or" to "and" for clarification because automatic pressure relief valves enable both core spray and LPCI mode of RHR during a small pipe break in the event of HPCI failure. This is acceptable.

9. TS Bases on pages 3.5/4.5-14 and 3.5/4.5-14b

Delete Ref. 5 from the Bases since it is no longer applicable. Also, Reference 1 is changed to incorporate the new loss of coolant accident model (SAFER/GESTR-LOCA).

10. Figure 3.5-1

Add new MAPLHGR curves for new fuel types BD300C and BD316A and delete MAPLHGR curves for fuel types no longer in use. This is acceptable.

11. Figure 3.5-2
Add the statement "For flows greater than 100%, K_f = 1.0" to the figure to address the operating region defined by the ICF analysis. This is acceptable.

12. TS 3.6.H.3 on page 3.6/4.6-5a and TS Bases on page 3.6/4.6-13a

Delete the MAPLHGR reduction factor during single loop operation based on the SAFER/GESTR-LOCA analysis. This is acceptable since an approved method was used. Revise RBM upscale limit due to new RBM setpoint. Reduce the allowed duration of unrestricted SLO to 12 hours.

3.0 ENVIRONMENTAL CONSIDERATION

This amendment involves changes to license requirements with respect to the installation and use of facility components located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and there is no significant increase in individual or cumulative occupational radiation exposure. The Commission previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement nor environmental assessment need be prepared in connection with the issuance of this amendment.

4.0 CONCLUSIONS

We have reviewed the reports submitted for Quad Cities Unit 2 Cycle 10 operation with extended operating regions and equipment out-of-service. Based on this review we conclude that appropriate material was submitted and that the fuel design, nuclear design, thermal-hydraulic design, and transient and accident analyses are acceptable. The Technical Specification and License Condition changes submitted for this reload suitably reflect the necessary modifications for operation during this cycle.

Furthermore, the staff concluded, based upon considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security nor to the health and safety of the public.

5.0 REFERENCES

1. Letter from J. A. Silady (CECo) to USNRC, dated March 28, 1988, "Quad Cities Station Unit 2 Proposed Technical Specification Amendment - Unit 2 Cycle 10 Reload NRC Docket No. 50-265."

2. GE document 23A5846, Revision 0, January 1988, "Supplemental Reload Licensing Submittal for Quad Cities Nuclear Power Station Unit 2, Reload 9, Cycle 10.

3. NEDC-31449, July 1987, "Extended Operating Domain and Equipment Out-of-Service for Quad Cities Nuclear Power Station Units 1 and 2."

4. NEDC-31345P, Revision 1, January 1988, "Quad Cities Nuclear Power Station Units 1 and 2 SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis." (Proprietary information. Not available in PDR)

5. Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A-6, Amendment 10."

6. GESTAR II, NEDE-24011, Revision 8, "General Electric Standard Application for Reactor Fuel."

7. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER models for the Evaluation of the Loss-of-Coolant Accident" Volume I, II, and III, General Electric Company, June 1984.

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Dated: June 17, 1988