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December 15, 1987

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Docket No.: 50-254

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Mr. L. D. Butterfield, Jr.
Nuclear Licensing Manager
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Butterfield:

SUBJECT: Quad Cities Nuclear Power Station, Unit 1 (TAC 66197)

The Commission has issued the enclosed Amendment No.103 to Facility Operating License No. DPR-29 for the Quad Cities Nuclear Power Station, Unit 1. This amendment is in response to your application dated September 18, 1987, as supplemented by October 13, 1987 and clarified by November 25, 1987.

In general, license conditions and Technical Specifications related to plant operating limits (e.g. LHGR, MCPR, MAPHLR, and RBM), operating domains including equipment out of service (e.g. SLO, ICF, FWTR and RVOOS), surveillance requirements, and affected bases are revised to reflect the new Cycle 10 fuel reload, transient, and accident analyses.

By letter dated November 25, 1987, Commonwealth Edison Company (CECo) has committed, independent of TS requirements, to monitor jet pump integrity during single loop operation by performing Core Plate Differential Pressure surveillance (using plant procedures). Furthermore, clipping of the rod block setpoint above 100% core flow will be accomplished by adjusting back-up circuity. However, to avoid potential confusion between the units, by using different RBM settings, CECo will defer the setpoint adjustments for both units until the next unit 2 refueling outage (Spring 1988). In the interim, operation in the ICF region will not be allowed.

Although, Technical Specifications without surveillance requirements for core stability monitoring during SLO were approved specifically for Cycle 10, the staff is convinced that instabilities could occur under certain circumstances (see enclosure 2). As such, CECo is requested to reevaluate the applicability of stability surveillance TS for all future operating cycles.

A copy of our related Safety Evaluation is also enclosed. The Notice of Issuance will be included individually in the Commission's Federal Register notices.

Sincerely,

/S/

Thierry M. Ross, Project Manager
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects

Enclosures:
1. Amendment No. 103 to License No. DPR-29
2. Safety Evaluation

cc w/enclosures:
See next page

PDIII-2:LA	PDIII-PM	OGC-Bethesda	PDIII-2:PD
LLuther	TRoss:bj		DMuller
12/07/87	12/7/87	12/10/87	12/15/87

Mr. L. D. Butterfield, Jr.
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. 50-254

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 103
License No. DPR-29

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

   A. The application for amendment by Commonwealth Edison Company (the licensee) dated September 18, 1987, as supplemented by October 13, 1987, and clarified November 25, 1987, 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. Additionally, paragraphs 3.C. and 3.K. of Facility Operating License No. DPR-29 are hereby deleted in their entirety; and paragraph 3.B. is amended to read as follows:

B. Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 103 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

C. (Deleted)

K. (Deleted)

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

FOR THE NUCLEAR REGULATORY COMMISSION

Daniel R. Muller

Daniel R. Muller, Director
Project Directorate III-2
Division of Reactor Projects - III,
IV, V and Special Projects
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: December 15, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 103

FACILITY OPERATING LICENSE NO. DPR-29

DOCKET NO. 50-254

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
(ii)	(ii)
1.0-5	1.0-5
1.1/2.1-7	1.1/2.1-7
Figure 2.1-1	Figure 2.1-1
Figure 2.1-3	Figure 2.1-3
3.2/4.2-14	3.2/4.2-14
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
3.5/4.5-11	3.5/4.5-11
3.5/4.5-12	3.5/4.5-12
3.5/4.5-13a	3.5/4.5-13a
3.5/4.5-14	3.5/4.5-14
3.5/4.5-15	3.5/4.5-15
Figure 3.5-1 (Sheets 1 thru 4)	Figure 3.5-1 (Sheets 1 thru 5)
3.6/4.6-5	3.6/4.6-5
3.6/4.6-5a	3.6/4.6-5a
	3.6/4.6-5b
3.6/4.6-13	3.6/4.6-13

TABLE OF CONTENTS (Cont'd)

II. Dose Equivalent I-131 - That concentration of I-131 (microcurie/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844, "Calculation of Distance Factors For Power and Test Reactor Sites."

JJ. Process Control Program (PCP) - Contains the sampling, analysis, and formulation determination by which solidification of radioactive wastes from liquid systems is assured.

KK. Offsite Dose Calculation Manual (ODCM) - Contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents, and in the calculation of gaseous and liquid effluent monitor alarm/trip setpoints.

LL. Channel Functional Test (Radiation Monitor) - Shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify operability including alarm and/ or trip functions.

MM. Source Check - The qualitative assessment of instrument response when the sensor is exposed to a radioactive source.

NN. Member(s) of the Public - Shall include all persons who are not occupationally associated with the plant. This category does not include employees of the utility, its contractors, or vendors. Also excluded from this category are persons who enter the site to service equipment or to make deliveries. This category does include persons who use portions of the site for recreational, occupational, or other purposes not associated with the plant.

OO. <u>DUAL LOOP OPERATION (DLO)</u> - Reactor power operation with both recirculation pumps running.

PP. <u>SINGLE LOOP OPERATION (SLO)</u> - Reactor power operation with one recirculation pump running.

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.

APRM Flow Reference Scram
and APRM Rod Block Settings

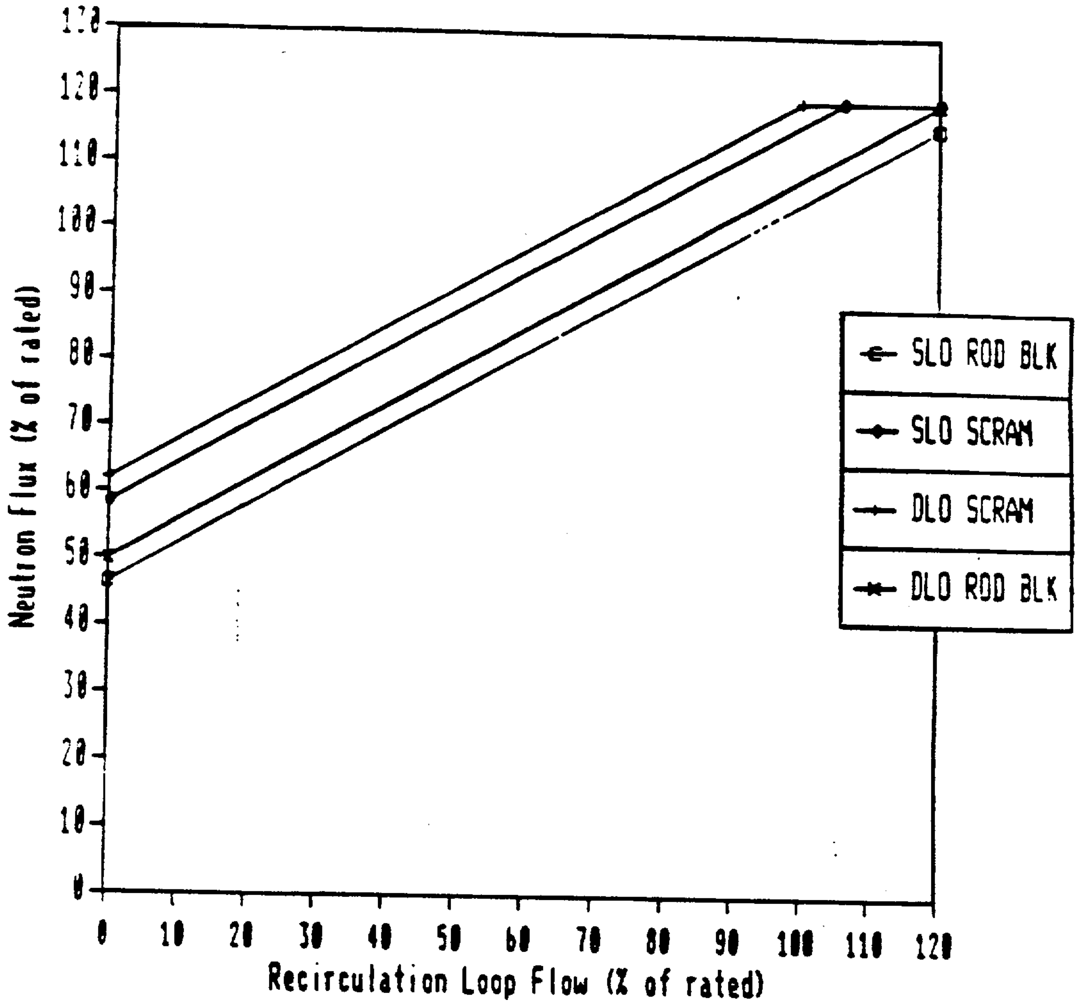


Figure 2.1-1

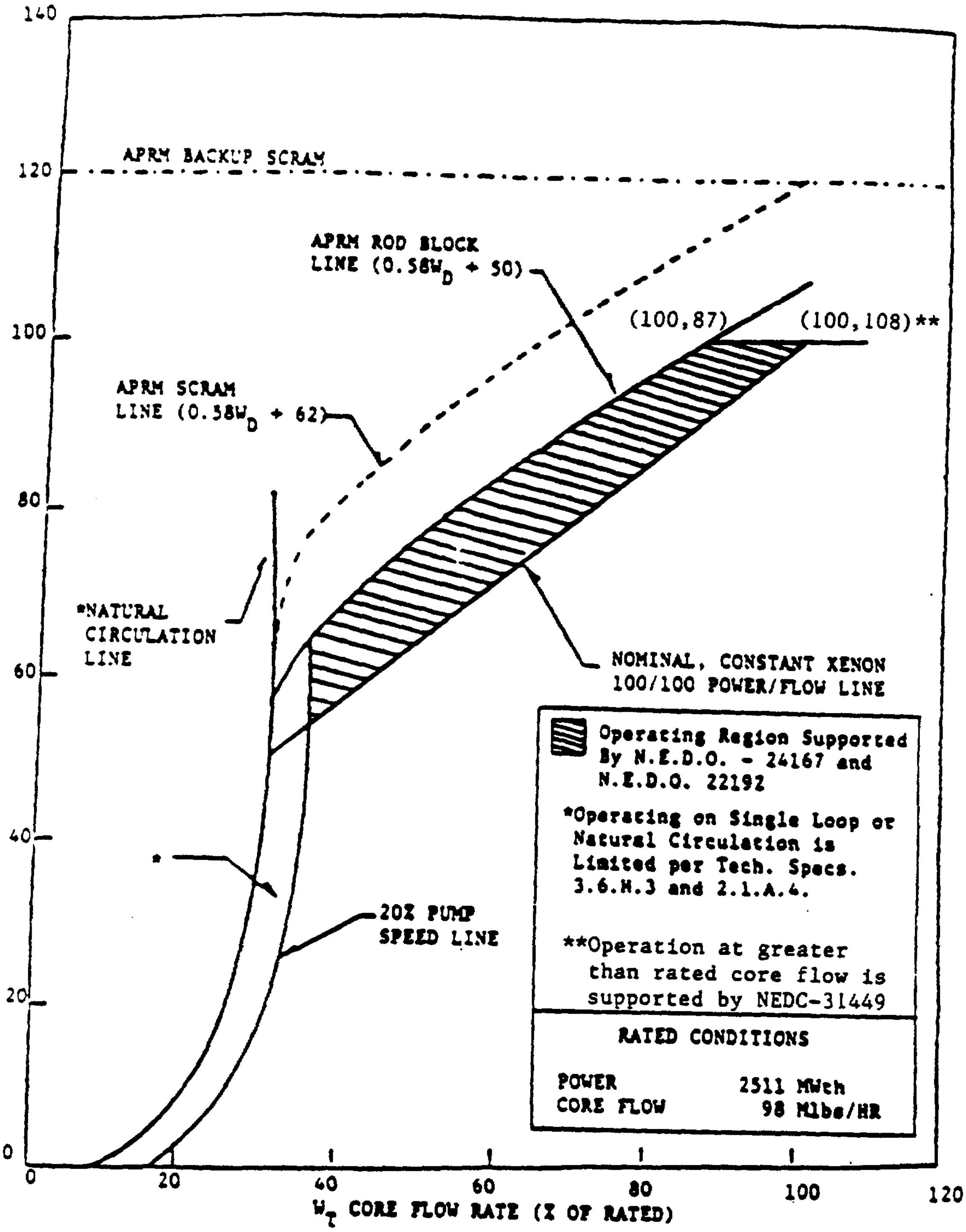


FIGURE 2.1-3
(SCHEMATIC)
APRM FLOW BIAS SCRAM RELATIONSHIP
TO NORMAL OPERATING CONDITIONS

Amendment No. 103

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)	$\leq$12/125 full scale
2	APRM downscale[7]	$\geq$3/125 full scale
1	Rod block monitor upscale (flow bias)[7]	$\leq 0.65W_D + 43$[2]
1	Rod block monitor downscale[7]	$\geq$3/125 full scale
3	IRM downscale[3] [8]	$\geq$3/125 full scale
3	IRM upscale[8]	$\leq$108/125 full scale
2[5]	SRM detector not in Startup position [4]	$\geq$2 feet below core centerline
3	IRM detector not in Startup position [8]	$\geq$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)	$\leq$ 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 GT/E 100CPS.

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

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

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

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.

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 0.71 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.

3. 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.

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

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.

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 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:

1.33 for $\tau_{AVE} \leq 0.71$ sec

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

$0.278\ \tau_{AVE} + 1.131$

for $0.71 \text{ sec} \leq \tau_{AVE} \leq 0.86$ 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.

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

This specification assures that adequate emergency cooling capability is available whenever irradiated fuel is in the reactor vessel.

Based on the loss-of-coolant analytical methods described in General Electric Topical Report NEDC-31345P core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact, to limit cladding metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. Under these Limiting Conditions of operation, increased surveillance testing of the remaining ECCS systems provides assurance that adequate cooling of the core will be provided during a loss-of-coolant accident.

Core spray distribution has been shown, in full-scale tests of systems similar in design to that of Quad-Cities 1 and 2, to exceed the minimum requirements by at least 25%. In addition, cooling effectiveness has been demonstrated at less than half the rated flow in simulated fuel assemblies with heater rods to duplicate the decay heat characteristics of irradiated fuel. The accident analysis is additional conservative in that no credit is taken for spray cooling of the reactor core before the internal pressure has fallen to 90 psig.

The LPCI mode of the RHR system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel cladding temperature. The LPCI mode of the RHR system in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.05 ft^2 up to and including 4.26 ft^2, the latter being the double-ended recirculation line break with the equalizer line between the recirculation loops closed without assistance from the high-pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 3. Using the results developed in this reference, the repair period is found to be less than

Should the loss of one RHR pump occur, a nearly full complement of core and containment cooling equipment is available. Three RHR pumps in conjunction with the core spray subsystem will perform the core cooling function. Because of the availability of the majority of the core cooling equipment, which will be demonstrated to be operable, a 30-day repair period is justified. If the LPCI mode of the RHR system is not available, at least two RHR pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. RHR Service Water

The containment cooling mode of the RHR system is provided to remove heat energy from the containment in the event of a loss-of-coolant accident. For the flow specified, the containment long-term pressure is limited to less than 8 psig and is therefore more than ample to provide the required heat-removal capability (reference SAR Section 5.2.3.2).

The Containment Cooling mode of the RHR System consists of two loops. Each loop consists of 1 Heat Exchanger, 2 RHR Pumps, and the associated valves, piping, electrical equipment, and instrumentation. The "B" loop on each unit contains 2 RHR Service Water Pumps. During the period from November 24, 1981, to July 1, 1982, the "A" loop on each unit may utilize the "A" and "B" RHR Service Water Pumps from Unit 2 via a cross-tie line. After July 1, 1982, each "A" loop will contain 2 RHR Service Water Pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. 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 provided the appropriate MAPLHGR reduction factor is applied to assure compliance with the 2200°F PCT limit. Loss of more than one relief valve 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.

H. Condensate Pump Room Flood Protection

See Specification 3.5.H

I. Average Planar LHGR

This specification assures that the peak cladding temperature following the postulated design-basis loss-of-coolant accident will not exceed the 2200°F limit specified in the 10 CFR 50, Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average heat-generation rate of all the rods of a fuel assembly at any axial location and is only secondarily dependent on the rod-to-rod power distribution within an assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the limit. The maximum average planar LHGR's shown in Figure 3.5-1 are based on calculations employing the models described in Reference 2.

The Average Planar Linear Heat Generation Rate (APLHGR) also serves a secondary function which is to assure fuel rod mechanical integrity.

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

The MCPR Operating Limit reflects an increase of 0.03 over the most limiting transient to allow continued operation with one feedwater heater out of service.

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.

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. "SAFER/GESTR-LOCA Loss of Coolant Analysis for QuadCities Nuclear Power Station Units 1 & 2" NEDC-31345P.*

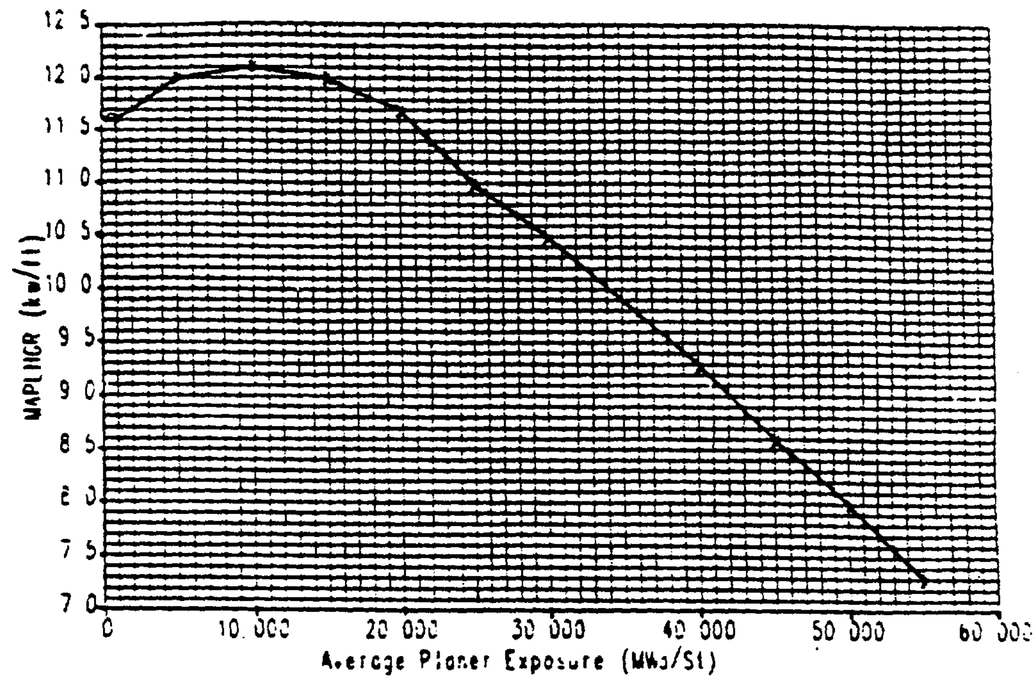
2. "Generic Reload Fuel Application," NEDE-24011-P-A**

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 Planer Exposure
Fuel Type Barrier LTA



MAPLHGR Vs. Average Planer Exposure
Fuel Types P8DRB265L/P8DGB265L

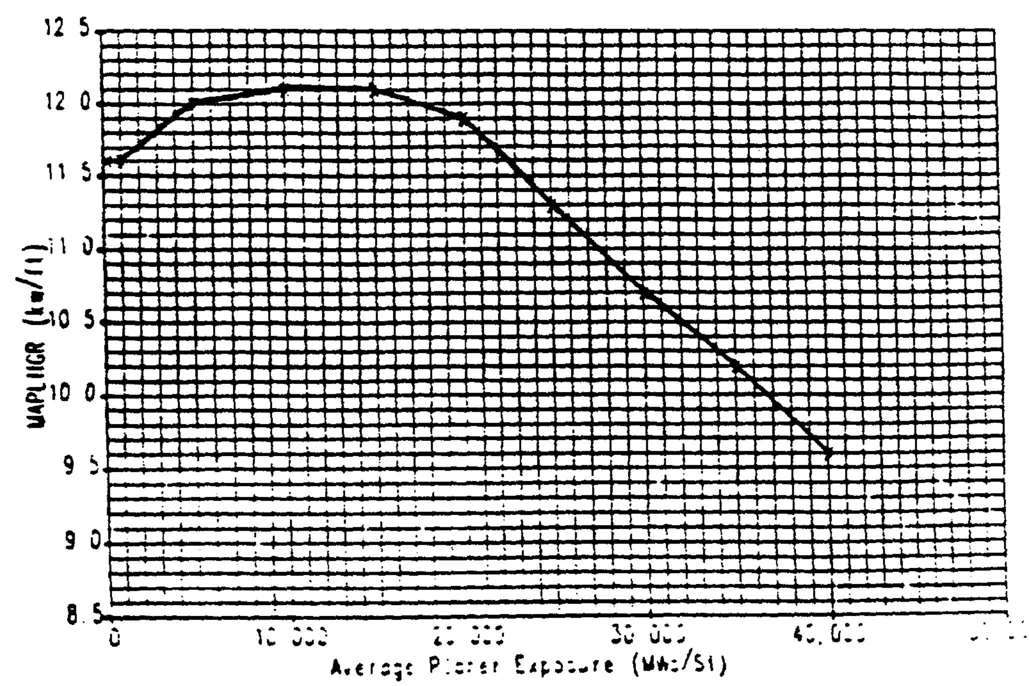
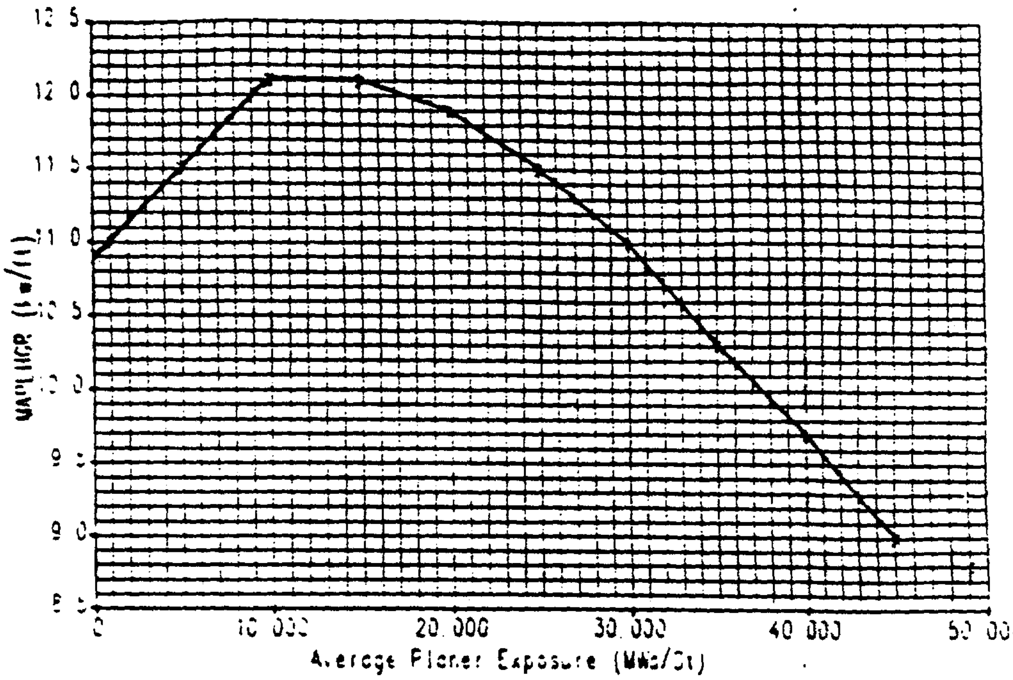


Figure 3.5-1 (Sheet 1 of 5)

MAPLHGR Vs Average Planer Exposure
Fuel Type BP8DRB299



MAPLHGR Vs Average Planer Exposure
Fuel Type P8DRB282

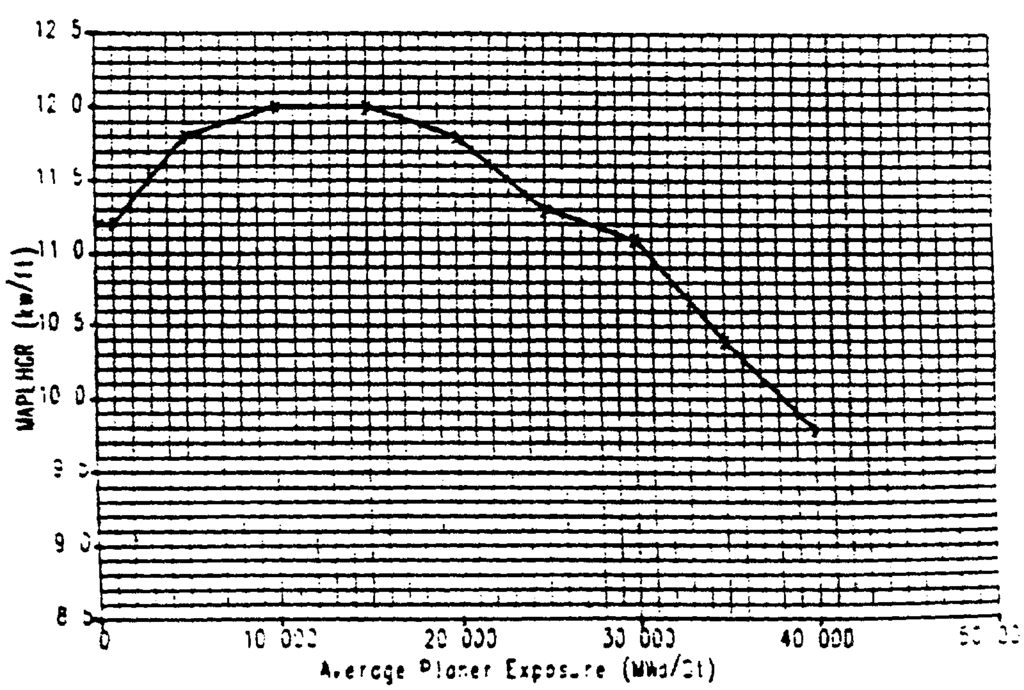
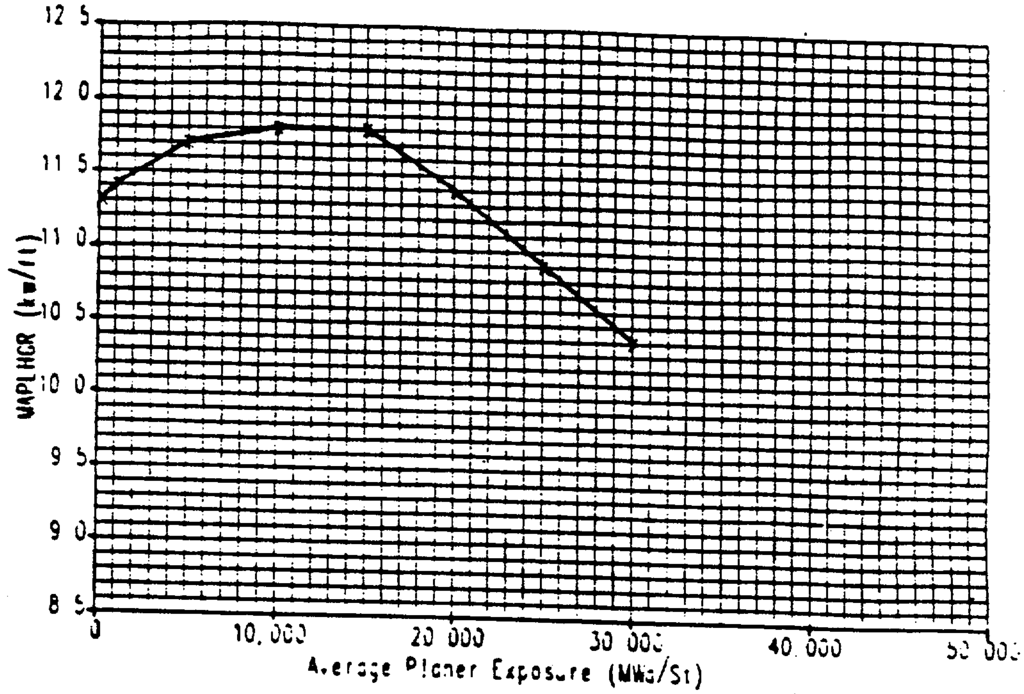


Figure 3.5-1 (Sheet 2 of 5)

MAPLHGR Vs Average Planer Exposure
Fuel Type P8DRB239



MAPLHGR Vs. Average Planer Exposure
Fuel Types P8DRB265H/BP8DRB265H

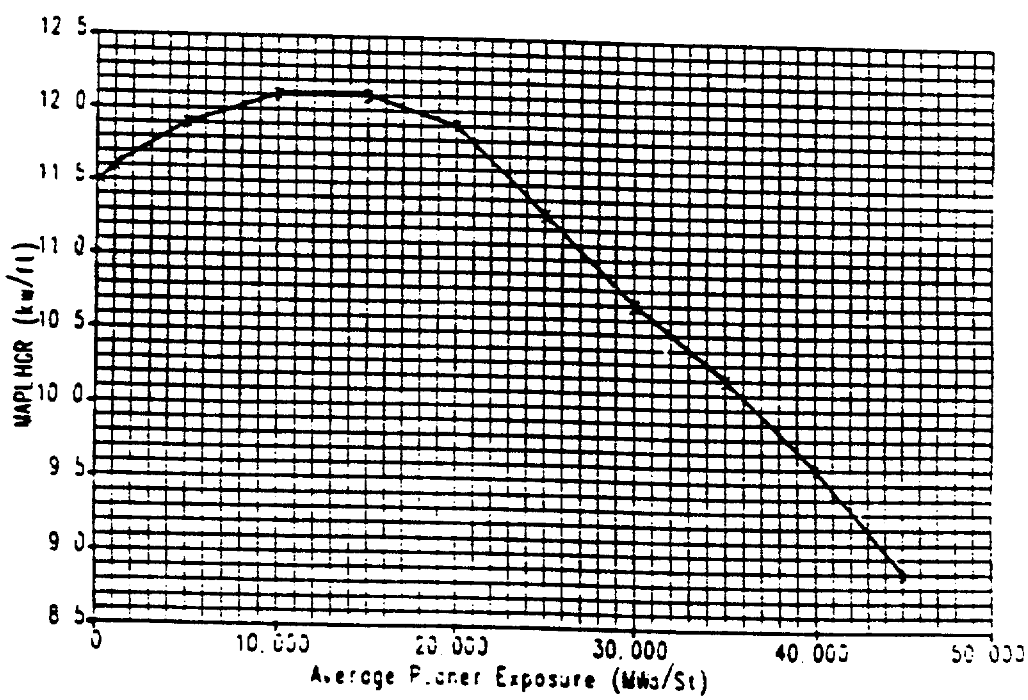
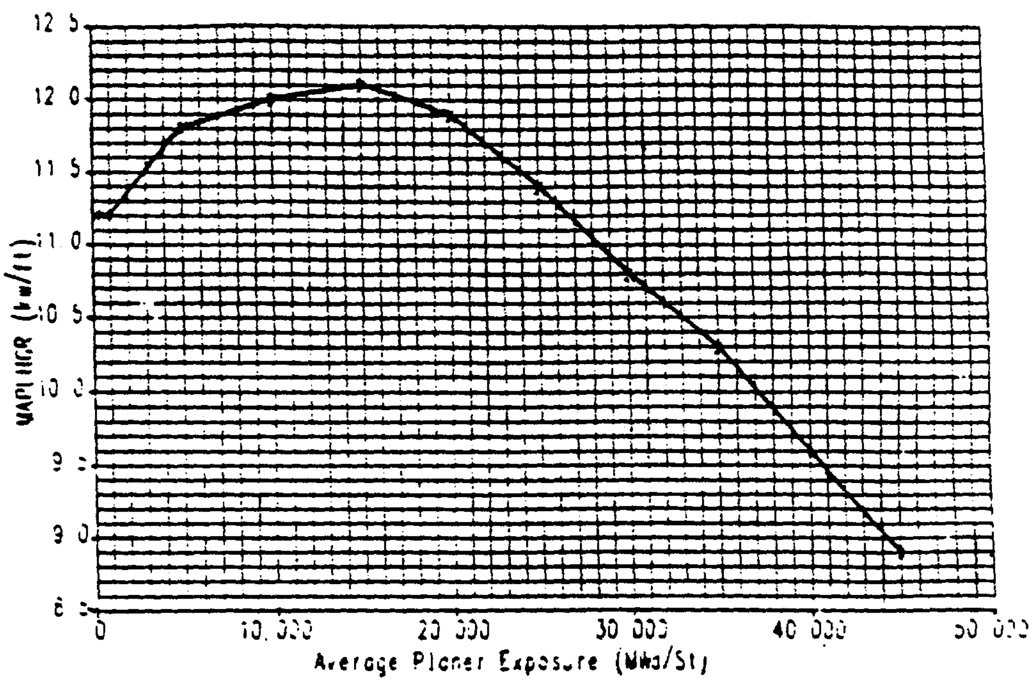


Figure 3.5-1 (Sheet 3 of 5)

MAPLHGR Vs Average Planer Exposure
Fuel Type BP8DRB282



MAPLHGR Vs Average Planer Exposure
Fuel Type BP8DRB283H

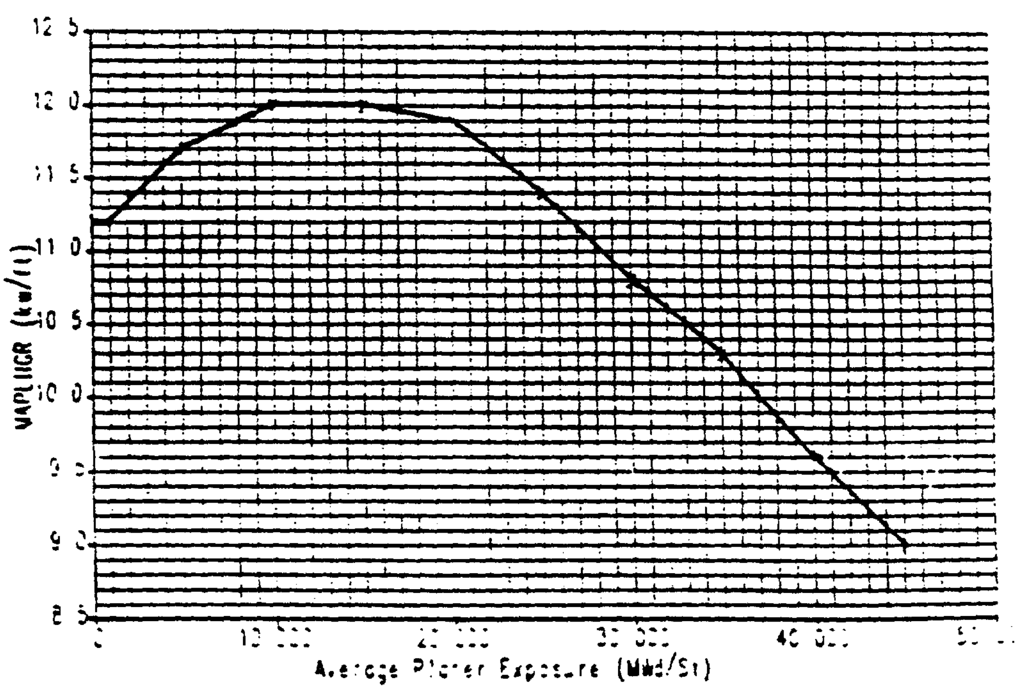
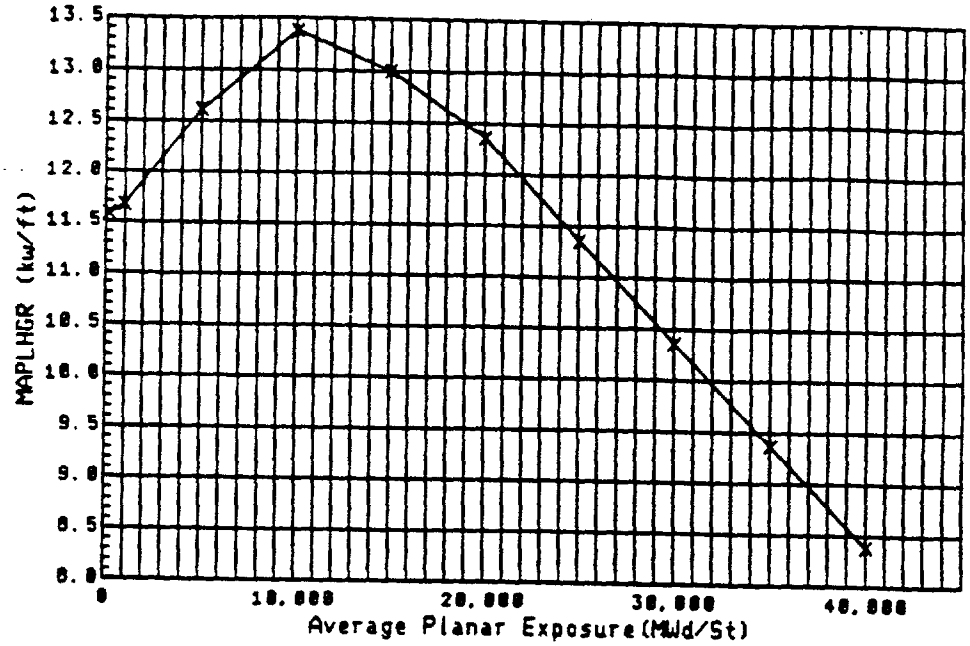


Figure 3.5-1 (Sheet 4 of 5)

MAPLHGR VS. Average Planar Exposure
Fuel Type BD300A



MAPLHGR VS. Average Planar Exposure
Fuel Type BD300B

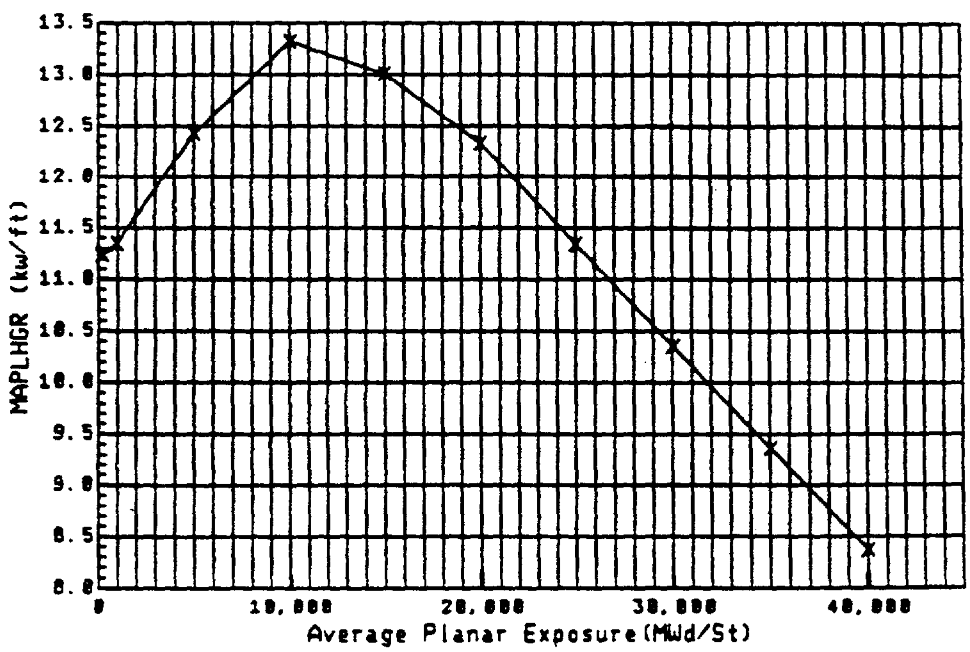


Figure 3.5-1 (Sheet 5 of 5)

G. Jet Pumps

1. Whenever the reactor is in the Startup/Hot Standby or Run modes, all jet pumps shall be intact, and all operating jet pumps shall be operable. If it is determined that a jet pump is inoperable, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

2. Flow indication from each of the 20 jet pumps shall be verified prior to initiation of reactor startup from a cold shutdown condition.

3. The indicated core flow is the sum of the flow indication from each of the 20 jet pumps. If flow indication failure occurs for two or more jet pumps, immediate corrective action shall be taken. If flow indication for all but one jet pump cannot be obtained within 12 hours, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

H. Recirculation Pump Flow Limitations

1. Whenever both recirculation pumps are in operation, pump speeds shall be maintained within 10% of each other when power level is greater than 80% and within 15% of each other when power level is less than 80%.

2. If Specification 3.6.H.1 cannot be met, one recirculation pump shall be tripped.

G. Jet Pumps

1. Whenever there is recirculation flow with the reactor in the Startup/Hot Standby or Run modes, jet pump integrity and operability shall be checked daily by verifying that the following two conditions do not occur simultaneously:

   a. The recirculation pump flow differs by more than 10% from the established speed-flow characteristics.

   b. The indicated total core flow is more than 10% greater than the core flow value derived from established core plate DP-core flow relationships.

2. Additionally, when operating with one recirculation pump with the equalizer valves closed, the diffuser to lower plenum differential pressure shall be checked daily, and the differential pressure of any jet pump in the idle loop shall not vary by more than 10% from established patterns.

3. The baseline data required to evaluate the conditions in Specifications 4.6.G.1 and 4.6.G.2 will be acquired each operating cycle.

H. Recirculation Pump Flow Limitations

Recirculation pumps speed shall be checked daily for mismatch.

3. During 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 ≤ .58WD + 58.5

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

      T.S. 2.1.B;
      S ≤ .58WD + 46.5

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

      T.S. 3.2.C (Table 3.2-3); *
      APRM UPSCALE ≤ (.58WD + 46.5) FRP/MFLPD

      * In the event that MFLPD excedes 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 ≤ .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.

I. Shock Suppressors (Snubbers)

1. During all modes of operation except Shutdown and Refuel, all snubbers listed in Table 3.6-1 shall be operable except as noted in 3.6.1.2 following.

2. From and after the time that a snubber is determined to be inoperable, continued reactor operation is permissible during the succeeding 72 hours only if the snubber is sooner made operable.

3. If the requirements of 3.6.1.1 and 3.6.1.2 cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

4. If a snubber is determined to be inoperable while the reactor is in the Shutdown or Refuel mode, the snubber shall be made operable prior to reactor startup.

5. Snubbers may be added to safety-related systems without prior license Amendment to Table 3.6-1 provided that a revision to Table 3.6-1 is included with the next license amendment request.

I. Shock Suppressors (Snubbers)

The following surveillance requirements apply to all snubbers listed in Table 3.6-1.

1. Visual inspections shall be performed in accordance with the following schedule utilizing the acceptance criteria given by Specification 4.6.1.2.

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ±25%
1	12 months ±25%
2	6 months ±25%
3,4	124 days ±25%
5,6,7	62 days ±25%
≥8	31 days ±25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, 'accessible' or 'inaccessible' based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

G. Jet Pumps

Failure of a jet pump nozzle assembly holddown mechanism, nozzle assembly, and/or riser increases the cross-sectional flow area for blowdown following the postulated design-basis double-ended recirculation line break. Therefore, if a failure occurs, repairs must be made to assure the validity of the calculated consequences.

The following factors form the basis for the surveillance requirements:

1. A break in a jet pump decreases the flow resistance characteristic of the external piping loop causing the recirculation pump to operate at a higher flow condition when compared to previous operation.

2. The change in flow rate of the failed jet pump produces a change in the indicated flow rate of that pump relative to the other pumps in that loop. Comparison of the data with a normal relationship or pattern provides the indication necessary to detect a failed jet pump.

3. The jet pump flow deviation pattern derived from the diffuser to lower plenum differential pressure readings will be used to further evaluate jet pump operability in the event that the jet pumps fail the tests in Sections 4.6.G.1 and 2.

Agreement of indicated core flow with established core plate DP-core flow relationships provides the most assurance that recirculation flow is not bypassing the core through inactive or broken jet pumps. This bypass flow is reverse with respect to normal jet pump flow. The indicated total core flow is a summation of the flow indications for the 20 individual jet pumps. The total core flow measuring instrumentation sums reverse jet pump flow as though it were forward flow. Thus, the indicated flow is higher than actual core flow by at least twice the normal flow through any backflowing pump. Reactivity inventory is known to a high degree of confidence so that even if a jet pump failure occurred during a shutdown period, subsequent power ascension would promptly demonstrate abnormal control rod withdrawal for any power-flow operating map point.

A nozzle-riser system failure could also generate the coincident failure of a jet pump body; however, the converse is not true. The lack of any substantial stress in the jet pump body makes failure impossible without an initial nozzle riser system failure.

H. Recirculation Pump Flow Limitations

The LPCI loop selection logic is described in the SAR, Section 6.2.4.2.5. For some limited low probability accidents with the recirculation loop operating with large speed differences, it is possible for the logic to select the wrong loop for injection. For these limited conditions, the core spray itself is adequate to prevent fuel temperatures from exceeding allowable limits. However, to limit the probability even further, a procedural limitation has been placed on the allowable variation in speed between the recirculation pumps.

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. 103 TO FACILITY OPERATING LICENSE NO. DPR-29

COMMONWEALTH EDISON COMPANY

AND

IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES NUCLEAR POWER STATION, UNIT 1

DOCKET NO. 50-254

1.0 INTRODUCTION

By letter from J. A. Silady, Commonwealth Edison Company (CECo, the licensee), to T. Murley, NRC, dated September 18, 1987 (Ref. 1), as supplemented by October 13, 1987 (Ref. 5), Technical Specification (TS) changes were proposed for the operation of Quad Cities Station Unit 1 for Cycle 10 (QC1C10) with a reload using General Electric (GE) manufactured fuel assemblies and GE analyses and methodologies. Enclosed were the requested changes and reports (including Reference 2 through 4) discussing the reload and analyses done to support and justify Cycle 10 operation including an increased flow operating region, equipment out of service and single loop operation.

The reload for Cycle 10 is in general a normal reload with no unusual core features or characteristics. TS changes are few and primarily related to Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) and Linear Heat Generation Rate (LHGR) limits for the new fuel and 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 newer 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 allowed operating region on the reactor power-flow map via an increased core flow (ICF) extension. Extended Load Line Limit Analysis (ELLLA) and associated TS have previously been approved for Quad Cities 1.

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. CECo has also proposed removing restrictions for SLO from the license and incorporating limits for such operation in the TS, and removing license restrictions for coastdown and concommitant FFWTR.

On November 25, 1987 (Ref. 11), following a series of conference calls with NRC staff, CECo clarified references 1 and 2 with additional information (on rod block setpoint clipping and jet pump integrity surveillance) and editorial change of the Automatic Depressurization System TS to assure a more conservative interpretation of their meaning . The previous CECo Safety Evaluation and No Significant Hazards Consideration analysis (see Ref. 1) encompasses this additional submittal. Consequently CECo's application was not renoticed in the Federal Register with Reference 11 due to the explanatory and editorial nature of this clarifying document.

2.0 EVALUATION

2.1 Reload Description

The QC1C10 reload will retain 524 BP8x8R and P8x8R GE fuel assemblies from the previous cycle and add 200 new GE8x8EB fuel assemblies. The reload is based on a previous end of cycle core nominal average exposure of 21.1 GWD/ST and Cycle 10 end of cycle exposure of 22.2 GWD/ST. The loading will be a conventional scatter pattern with low reactivity fuel on the periphery. This loading is acceptable.

2.2 Fuel Design

The new fuel for Cycle 10 is the GE extended burnup fuel GE8x8EB. The fuel designations are BD 300A and BD 300B. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Refs. 6 and 7). The specific descriptions of this fuel have been submitted in Amendment 18 to GESTAR II, but since this amendment has not as yet been accepted, the fuel description has also been presented for QC1C10 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 (see Section 2.5). 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 the 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 7, 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). This LHGR has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Ref. 6). (See the referrals in Reference 6 to References 18 and 19. These references are responses to questions and presentations relating to the GE8x8EB fuel which provide information on the 14.4 kW/ft LHGR.) This LHGR is acceptable for the fuel in QC1C10.

2.3 Nuclear Design

The nuclear design for QC1C10 has been performed by GE with the approved methodology described in GESTAR II (Ref. 7, Section 3, Vol. 1). 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 0.0132 delta k at both BOL and at the exposure of minimum shutdown margin respectively, thus fully meeting the required 0.0038 delta k. The Standby Liquid Control System also meets shutdown requirements with a shutdown margin of 0.041 delta k. Since these and other QC1C10 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 QC1C10 has been performed by GE with the approved methodology described in GESTAR II (Ref. 7, Section 4, Vol. 1 and Section 2, Vol. 2) and the results are given in the GE reload report (Ref. 2). The parameters used for the analyses are those approved in GESTAR II for the Quad Cities class BWR 3. The GEMINI system of methods (approved in Ref. 8, See Ref. 7, Vol. 2) was used for relevant transient analyses.

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) and Load Rejection Without Bypass (LRWBP). The analyses of these events for QC1C10, using the standard, approved (Ref. 7) GEMINI ODYN Option A and B approach for pressurization transients in standard and extended operating regions and with analyzed equipment out of service combinations (see Section 2.6) provide new Cycle 10 TS values of OLMCPR as a function of average scram time. For all standard operating conditions LRWBP is controlling at both option A and B limits, giving OLMCPR values of 1.33 and 1.28 respectively. With the selected rod block setting of 108% the RWE is not limiting. However, to accommodate the extended and equipment out-of-service conditions discussed in Section 2.6 the OLMCPR has been analyzed (Ref. 3) for those conditions also. This has resulted in an increase to 1.37 for Option A and 1.33 for Option B. This is determined primarily by limits associated with the feedwater heater out-of-service analyses. The Option B value includes an increase of 0.01 to accommodate the possibility of an increase in measured scram time (to 20 percent insertion) from the standard 0.68 to 0.71. These OLMCPR results are reflected in TS changes. Approved methods (Ref. 7) were used to analyze these events (and others which could be limiting) and the analyses and results are acceptable and fall within expected ranges.

At the request of CECo, 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 QC1C10. The result is 0.59. This indicative of a stable core since it is substantially less than the accepted value of 0.8 (for approved GE methods). Further, previous cores for Quad Cities 2 have a history of stable operation and low calculated stability decay ratios. This is sufficient to require no two recirculation loop operation stability surveillance. However, the staff has generally required surveillance for single loop operation which will be permitted for QC1C10 (see Section 2.6).

The licensee has concluded that stability monitoring surveillance provisions are not required for SLO technical specifications since it is demonstrably stable. USNRC Generic Letter 86-09 is cited to justify this position for BWR/3s. While the staff agrees that this position is justifiable for operating Cycle 10, we do not agree that Generic Letter 86-09 supports the approval of permanent SLO without the inclusion of stability surveillance requirements in low flow operating regions. Recent operating experience at a foreign BWR-3 plant has shown that instabilities do occur in BWR-3 reactor types under some circumstances of core design and operating conditions. While the staff accepts the proposed SLO technical specifications without surveillance provisions for Cycle 10, the licensee is requested to reevaluate the need for stability surveillance specifications in future operating cycles based on the stability characteristics of the proposed operation. This determination can be made by the licensee based on calculations or other evidence which demonstrates that the low stability decay ratio is being maintained in future reload core designs.

2.5 Transient and Accident Analyses

The transient and accident analysis methodologies used for QC1C10 are described and NRC approval indicated in GESTAR II (Ref. 7, Vol. 2). The GEMINI system of methods (Refs. 6 and 8) option was used for transient analyses. The limiting MCPR events for QC1C10 are indicated in Section 2.4. The core wide transient analysis methodologies and results are acceptable and fall within expected ranges.

The RWE was analyzed on a plant and cycle specific basis (as opposed to the statistical approach) and a rod block setpoint of 108% was selected to provide an OLMCPR of 1.23 for all fuel types. This is less limiting than the core wide events. The mislocated assembly event is not analyzed for reload cores on the basis of NRC approved (see Reference S.2-59 of Ref. 7) studies indicating the small probability of an event exceeding MCPR limits. The fuel assembly misorientation was analyzed by GE with standard NRC approved methods and this OLMCPR was also less limiting than the other events. The local transient event analyses are thus acceptable.

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, extended and equipment out-of-service conditions well under required limits. These are acceptable methodologies and results.

Since Banked Position Withdrawal Sequence and rod patterns are used for Quad Cities 1, a cycle specific control rod drop accident analysis is not required. The basis for this position and NRC approval is presented in Amendment 9 in Reference 7.

The LOCA analyses for QC1C10 were performed using the SAFER/GESTR-LOCA methodology. This methodology (Ref. 9 and 7) has been approved by the staff and used and approved in several recent reload applications (e.g., Duane Arnold Cycle 9).

In Reference 9 the staff has specified the necessary conditions for demonstrating applicability of the SAFER/GESTR-LOCA methodology. These conditions are:

1. Calculation of a sufficient number of plant specific PCT points based on both nominal input values and Appendix K values to verify the shape of the PCT curves versus break size.

2. Confirmation that plant specific operating parameters have been bounded by the models and inputs used in the generic calculations.

3. Confirmation that the plant specific ECCS configuration is consistent with the referenced plant class ECCS configuration.

The licensee has reported the results of those analyses (Ref. 4) which are required to meet these conditions. Specifically, the analyses include break sizes from 0.05 ft^2 to the DBA recirculation suction line break (4.26 ft^2). Seven different break sizes were analyzed in conjunction with ECCS failure combinations. A total of 16 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 1 (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 calculated PCT is 828° F when nominal input values are used and 1382° F when Appendix K input values (plus adder) are used. The input parameters, the ECCS combination and the cases analyzed to establish the trend of PCT verse break size meet the staff requirements given above. 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 effects of extended or equipment out-of-service operation (Refs. 3 and 4). This included the full range discussed in Section 2.6. The change to peak clad 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 QC1C10 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. In addition, previously approved ELLLA conditions continued to be supported by the analyses for this cycle. 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 analyses are intended to provide a basis for the removal of Quad Cities 1 license restrictions and, for SLO TS additions.

For ICF the proposed operating region is extended to 108 percent core flow (up to 100 percent power), and the Safety Evaluation for this operating region (Ref. 3) 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 limiting transients and accidents examined for standard operating conditions were reexamined, with calculations at representative extreme operating points. The RWE analysis assumes that the Rod Block Monitor is clipped at 108 percent power so that the block conditions do not change above 100 percent flow. This clipping will be provided for by CECo (Ref. 11). The transient analyses were used to determine OLMCPR values for these operating conditions. (As discussed in Section 2.4 OLMCPR for QC1C10 is determined by the analysis of FWHOOS.) The LOCA examination concluded that the effects on MAPLHGR (and other limits) were insignificant compared to the large margin available. It was also concluded that core stability is possibly slightly affected but is well within established criteria. 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 accidents 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 1. This approved mode of operation bounds the license restriction on coastdown and feedwater heating which CECo proposes to remove (see Section 2.7).

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 result in setting the OLMCPR for QC1C10. The increased limit is caused primarily by changes in axial power distribution and resulting effectiveness of scram action. The conclusions for other aspects of affects or events (including loads, vibration and internal fatigue) are similar to that for ICF-FFWTR. The review concludes that operation with FWHOOS is acceptable for QC1C10. Operation for other cycles is also acceptable if the limits on MCPR presented in Reference 3, Section B.2.1 for this and other events are met.

For RVOOS the limiting pressurization event was evaluated with the most limiting relief valve out. The impact on MCPR is negligible. Standard sensitivity studies also show the effect of 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.

CECo proposes to remove the license condition on SLO for Quad Cities 1 and to provide appropriate TS additions covering needed changes and limits. The changes and accompanying analyses and staff review are in most respects completely similar to the changes approved by the staff for Quad Cities 2 (QC2) along with its Cycle 9 reload. That review is given in Reference 10. Only the highlights of the changes will be indicated here. Previously Quad Cities 1 had been allowed restricted SLO to less than 50 percent power. These restrictions are to be removed from the license and replaced by suitable TS. These TS are provided to (1) increase the safety limit MCPR and OLMCPR by 0.01 in SLO to account for increased core measurement uncertainties, (2) change the APRM scram and rod block and Rod Block Monitor flow biased setpoints to account for core vs measured flow changes, and (3) require the suction valve in the idle loop to be closed and isolated (and, as with present operating practice, the cross tie is to be closed). These changes are similar to those approved for QC2. The relevant events are analyzed by GE for Quad

Cities 1 in Reference 3. The 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 clad temperature well below limits. The stability situation for SLO has been previously discussed in Section 2.4. Similar to the operation approved for QC2 (Ref. 10) the core plate delta p noise surveillance will be incorporated in Unit 1 plant procedures (Ref. 11). The review of SLO operation has concluded that, as in the similar change for QC2, appropriate events, conditions analyses and methodologies have been used to examine SLO operation, that appropriate TS conditions and values have been proposed, and thus the removal of SLO restrictions from the license and proposed additions to the TS are acceptable.

2.7 Technical Specifications

The following TS changes have been proposed for Quad Cities 1 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 here. Two (3/4.5.D and 4.6.G) require further explanation. The TS will be followed by a similar listing of changes to a Definition, Bases, and administrative details. All of these TS and related changes suitably reflect the technical content approved in the preceding review and are acceptable. In addition, as previously discussed, the proposed removal of license requirements for coastdown, feedwater heating and SLO is also acceptable.

TS Changes

Figure 2.1-1

Appropriate lines giving APRM scram and rod block setpoint for SLO are added.

Figure 2.1-3

The approved ICF region is added to the power-flow map.

Table 3.2-3

The RBM trip level setting is increased to 108 percent as used in the RWE analysis.

3.3.C.5

The 20 percent scram insertion time is changed to 0.71 sec. corresponding to the ODYN B analysis.

3.5.D.2 and 4.5.D.4

This TS currently permits only limited (7 days) operation (during some conditions) with one of the relief valves of the automatic pressure relief system out of service. An interpretation of the original request would have permitted limited operation with all five out. At the staff request this has been clarified (see Ref. 11) to assure only two are allowed out. As previously discussed, one RVOOS has now been included in relevant analyses (e.g., LOCA). Thus the two RVOOS condtion is essentially the same as the previous one RVOOS condition. The phrase "and weekly thereafter" has been removed from 4.5.D.4 since it is redundant.

3.5.J

The approved 14.4 kW/ft LHGR limit for the extended burnup fuel has been added.

3.5.K

The approved OLMCPR values of 1.37 (Option A) and 1.33 (Option B) and the 0.71 sec Option B 20 percent scram time have replaced the previous values.

Figure 3.5-1 (Sheets 1-5)

The approved MAPLHGR values for the new fuel are provided.

4.6.G.1

The flow measurement has been changed from the power-flow to the core plate delta p -flow relationship because it has been shown to be more accurate. This change has previously been reviewed and approved for QC2.

3/4.6.H and 3.6.H.3

The title has been changed to reflect the new SLO TS which have been added (after SLO removal as a license restriction). The old 3.6.H.3 SLO has been removed and the new TS for approved SLO MCPR, APRM and RBM settings and value operation added. A new page, 3.6/4.6-5b has been added to accommodate the change.

Bases and Administrative Changes

Definitions OO and PP

Suitable definitions for Dual Loop Operation (DLO) and SLO have been added.

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The title change for 3.6/4.6H has been made.

B2.1

2511 MWt has been identified as the licensed maximum power.

B3.5.A

Changes the Reference and LOCA break size to reflect the new LOCA methodology.

B3.5.D

An "or" has been changed to "and" since both systems are enabled, and a paragraph has been added to discuss ADS with one valve out of service as now included in the LOCA analysis.

B3.5.I

Adds a paragraph about the mechanical design basis for MAPLHGR.

B3.5.K

Reflects approved operation with FWHOOS.

B.3.5 References

Change to 1. to reflect new LOCA model. Delete 5. since the change to GEMINI obsoletes this reference.

B3.6.G

Discusses delta p -flow relationship.

B3.6.H

Discusses new TS for SLO.

3.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.32 the Commission has determined that granting this amendment will have no significant impact on the environment (52 FR 47466).

4.0 CONCLUSIONS

The NRC staff has reviewed the reports submitted for the Cycle 10 operation of Quad Cities 1 with extended operating regions and equipment out-of-service. Based on this review the staff concludes 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 changes submitted for this reload suitably reflect the necessary modifications for operation in this cycle.

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

5.0 REFERENCES

1. Letter, and attachments, from J. A. Silady, CE, to T. Murley, NRC, dated September 18, 1987, "Quad Cities Station Unit 1 Proposed Technical Specification Amendment - Unit 1 Cycle 10 Reload", Attachments (1) Summary, (2) Significant Hazards Consideration, and (3) Proposed License and Technical Specification Changes.

2. GE Report 23A5831, Rev. 0, dated June 1987, "Supplemental Reload Licensing Submittal for Quad Cities Nuclear Power Station, Unit 1, Reload 9."

3. NEDC-31449, July 1987, Extended Operating Domain of Equipment Out-of-Service for Quad Cities ...."

4. NEDC-31345P, June 1987, "Quad Cities ... SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis."

5. Letter, and attachment, from J. A. Silady, CECo, to T. Murley, NRC, dated October 13, 1987, "... Supplemental Information ... Cycle 10 Reload."

6. Letter (and attachment) 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."

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

8. Letter (and attachment) from G. Lainas, NRC, to J. Charnley, GE, dated March 22, 1986, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-A, 'GE Generic Licensing Reload Report', Supplement to Amendment 11."

9. 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.

10. Letter, and attachments, from J. Zwolinski, NRC, to D. Farrar, CE, dated January 6, 1987, "Cycle 9 Reload and Single Loop Operation."

11. Letter, and attachments, from J. Silady, CECo, to T. Murley, NRC, dated November 25, 1987 "Supplemental Reload Licensing Information for Cycle 10."

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