

ATTACHMENT 3
(cont'd)

B. QUAD CITIES UNIT 2 CYCLE 10 LICENSE

AND

TECHNICAL SPECIFICATION CHANGES

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- C. Pursuant to the Act and 10 CFR Part 70, to receive, possess and use at any time, any byproduct, source and special nuclear materials as sealed neutron sources for reactor startup, sealed sources for reactor instrumentation and radiation monitoring equipment calibration, and as fission detectors in amounts required;
- Am. 36 D. Pursuant to the Act and 10 CFR Part 30, 40 and 70, to receive, possess and use in amounts as required any byproduct, source and special nuclear materials without restriction to chemical or physical form, for sample analysis or instrument and equipment calibration or associated with radioactive apparatus or components;
2/03/77
- Am. 41 E. Pursuant to the Act and 10 CFR Parts 30 and 70, possess, but not separate, such byproduct and special nuclear materials as may be produced by the operation of Quad Cities Nuclear Power Station, Unit Nos. 1 and 2.
1/30/78
3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Chapter I: Part 20, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70; is subject to all applicable provisions of the Act and to the rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:
- A. Maximum Power Level
Commonwealth Edison is authorized to operate Quad Cities unit No. 2 at power levels not in excess of 2511 megawatts (thermal).
- Am. 99 B. Technical Specifications
8/06/88
The Technical Specifications contained in Appendices A and B as revised through Amendment No. 99 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

- 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 2.1A and a control rod scram does not occur.

- 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:

$$S \leq (.58W_0 + 62)$$

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

where

S = setting in percent of rated power

W_0 = 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:

$$S \leq (.58W_0 + 62) \left[\frac{FRP}{MFLPD} \right]$$

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 Reference 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.

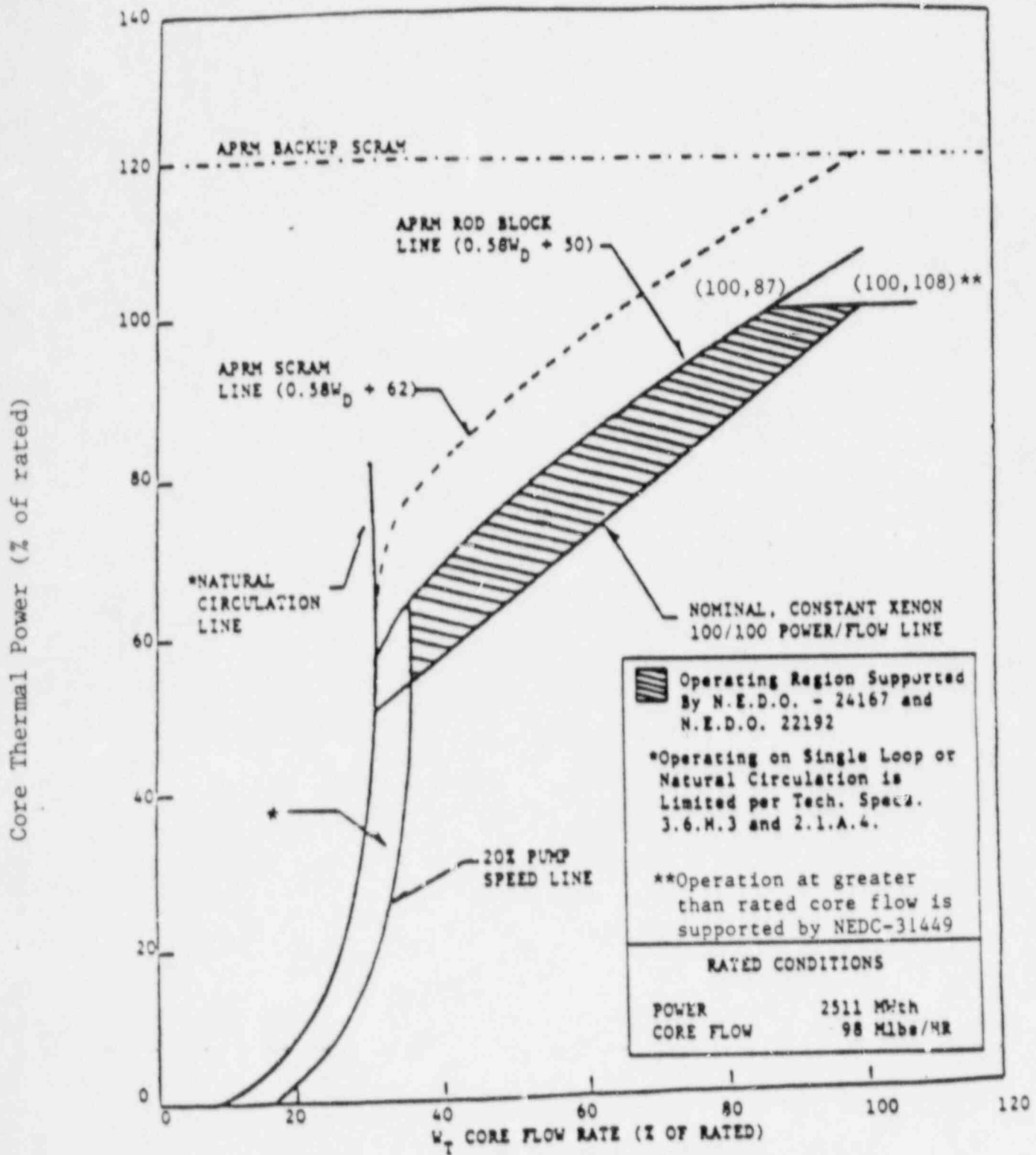


FIGURE 2.1-3
(SCHEMATIC)

APRM FLOW BIAS SCRAM RELATIONSHIP
TO NORMAL OPERATING CONDITIONS

Amendment No.

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]	$\leq [0.58W_0 + 50] \frac{FRP}{MFLPD}$ [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]	$\leq 0.65W_0 + 43$ [2] [10]
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]	≥ 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

- 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 ≥ 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 98×10^6 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.

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 $1\Delta k$. 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.

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%.

D. Control Rod Accumulators

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

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

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 BP0x8R. 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 $0.68 \text{ sec} \leq \tau_{AVE} \leq .86 \text{ sec}$

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 ODDYN 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 ODDYN 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.

QUAD-CITIES
DPR-30

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:

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

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 \bar{T}_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 \bar{T}_{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.

QUAD-CITIES
DPR-30

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.

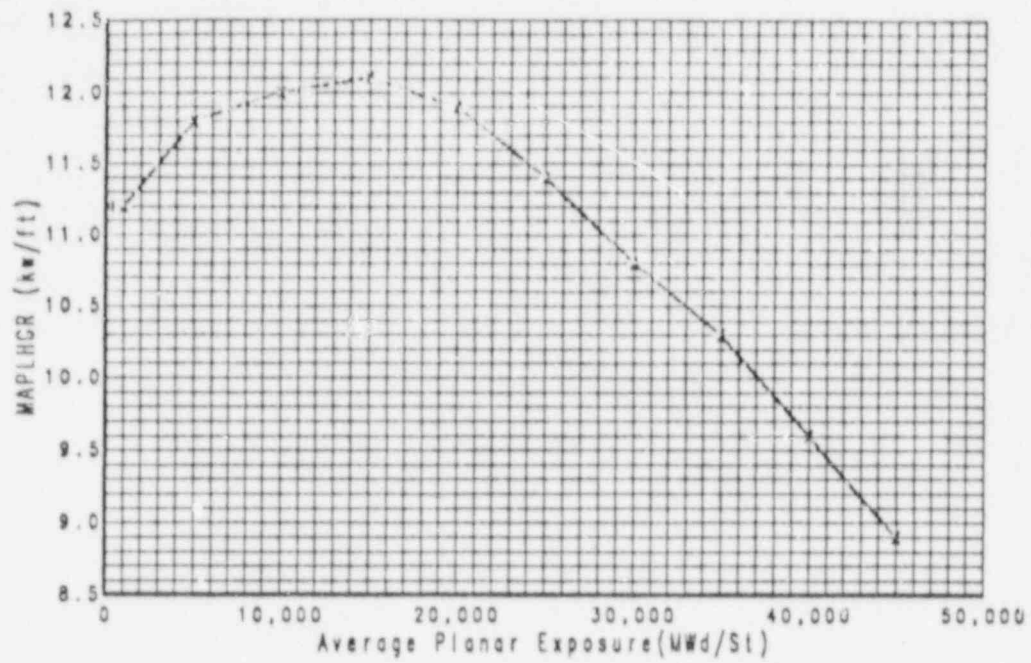
References

1. "Quad Cities Nuclear Power Station Units 1 & 2 SAFER/GESFR-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).

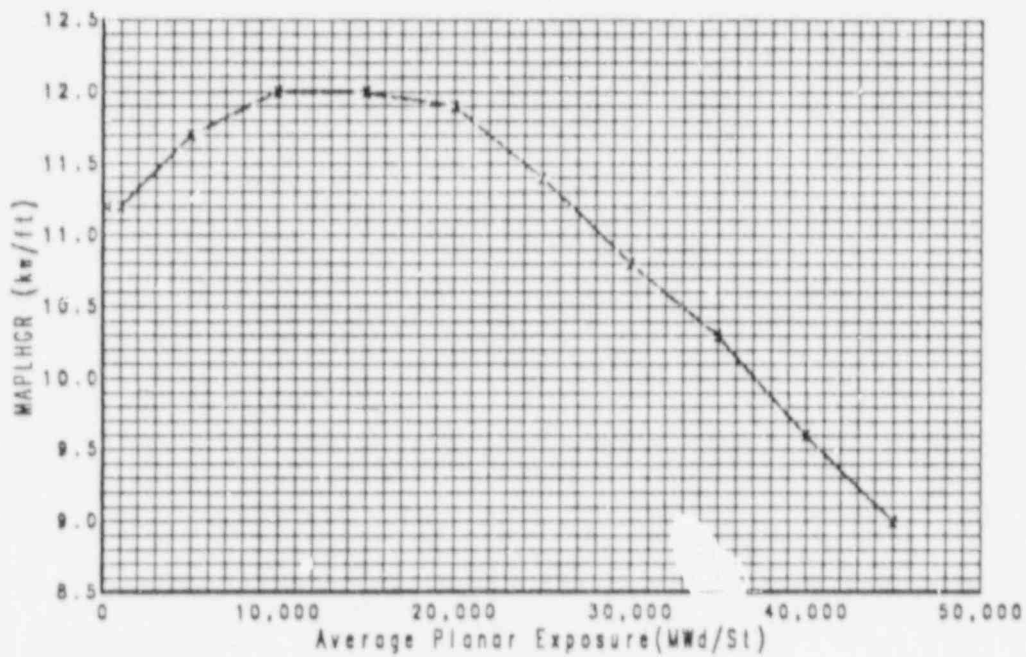
* Approved revision at time of plant operation.

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

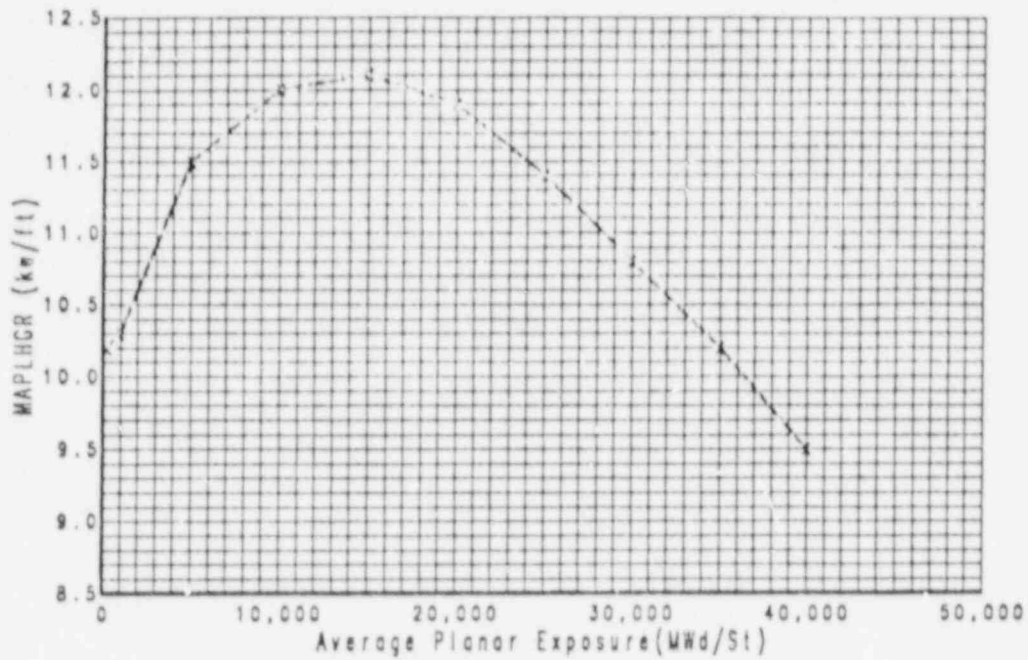
MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB282



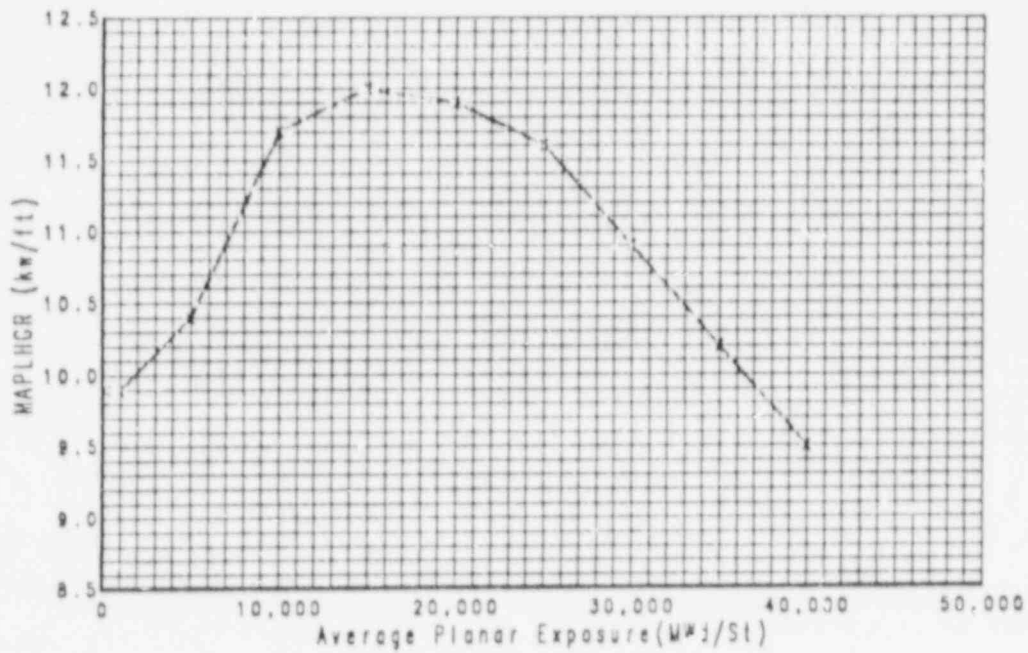
MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB283H



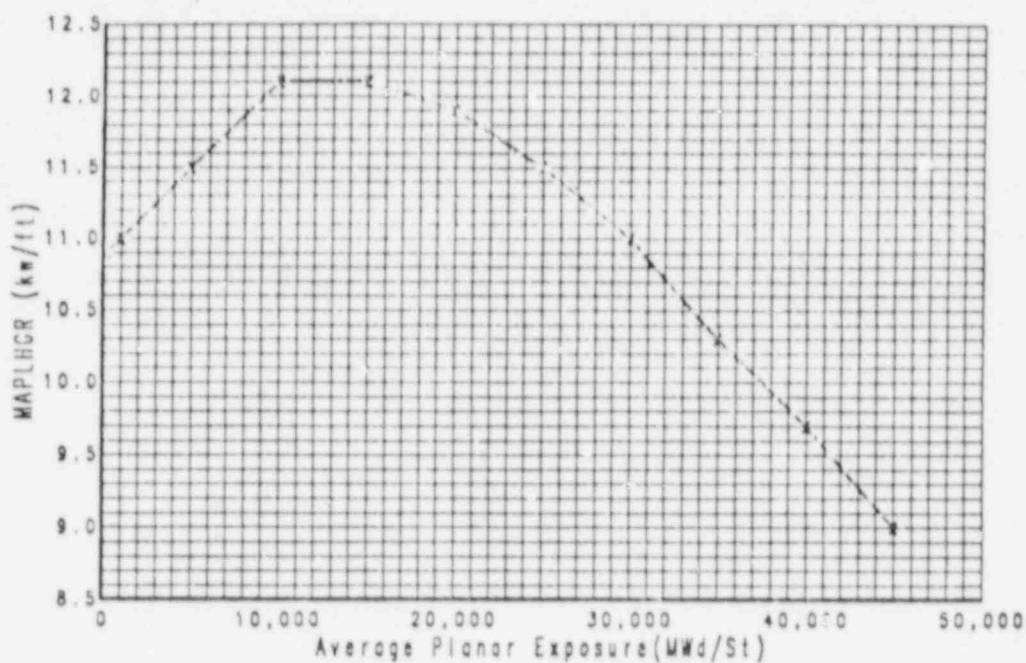
MAPLHGR VS. Average Planar Exposure
Fuel Type P8DGB263L



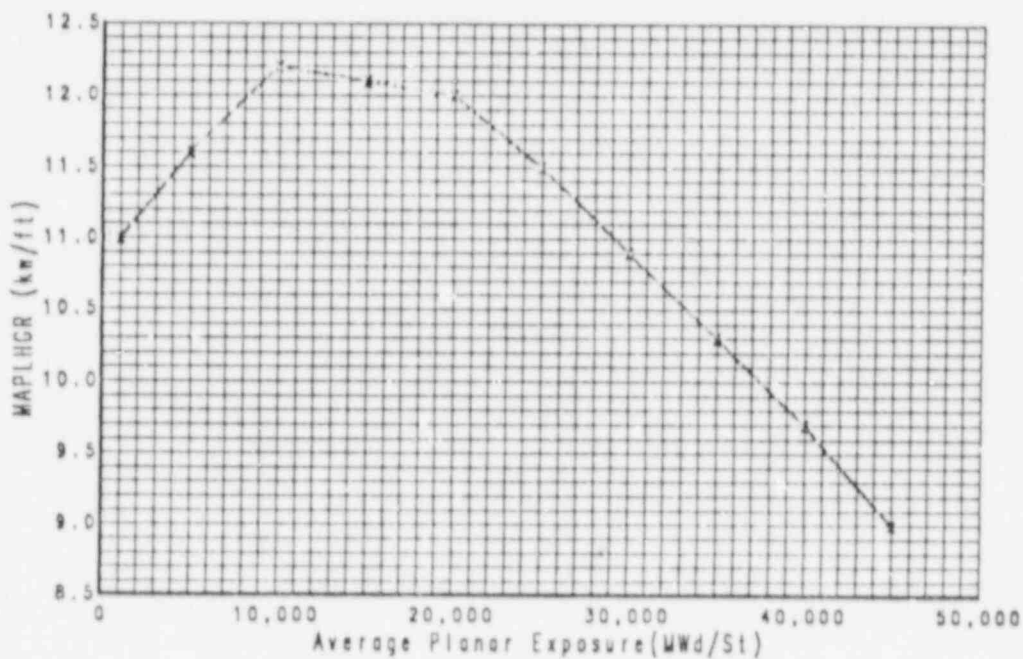
MAPLHGR VS. Average Planar Exposure
Fuel Type P8DGB298



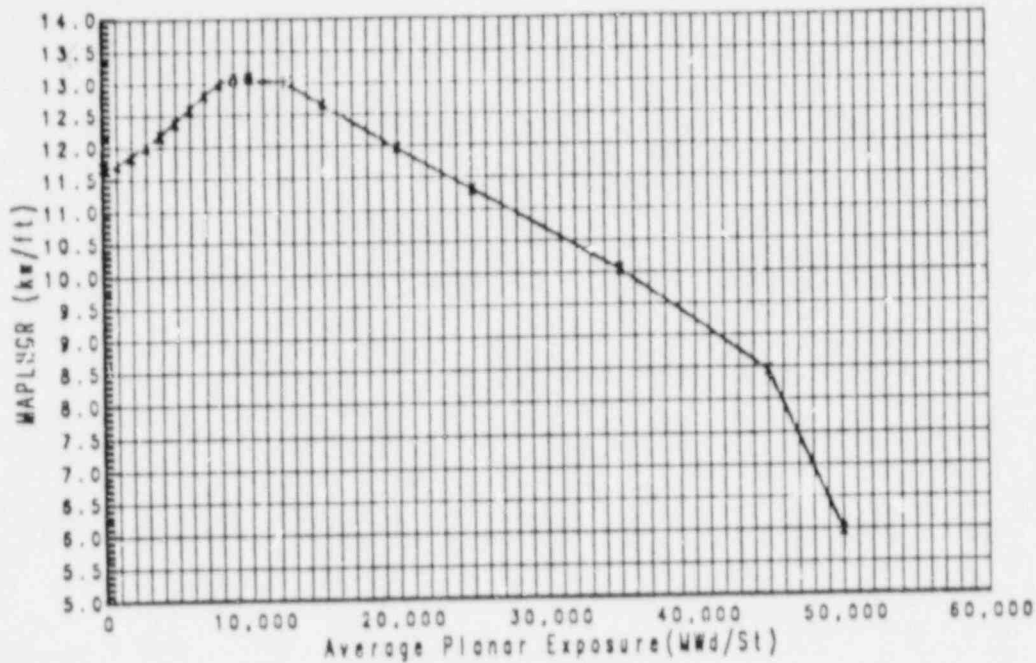
MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB299



MAPLHGR VS. Average Planar Exposure
Fuel Type BP8DRB299L



MAPLHGR VS. Average Planar Exposure
Fuel Type BD316A



MAPLHGR VS. Average Planar Exposure
Fuel Type BD300C



MAPLHGR VS. Average Planar Exposure
Fuel Types P8DRB265H, BP8DRB265H

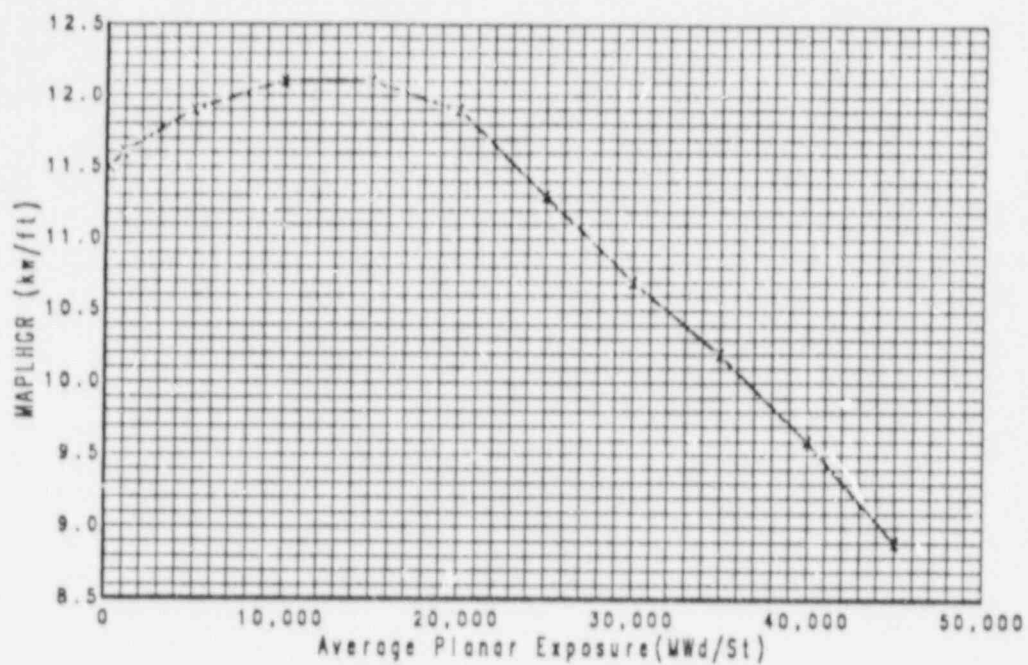


Figure 3.5-1
Sheet 5 of 5

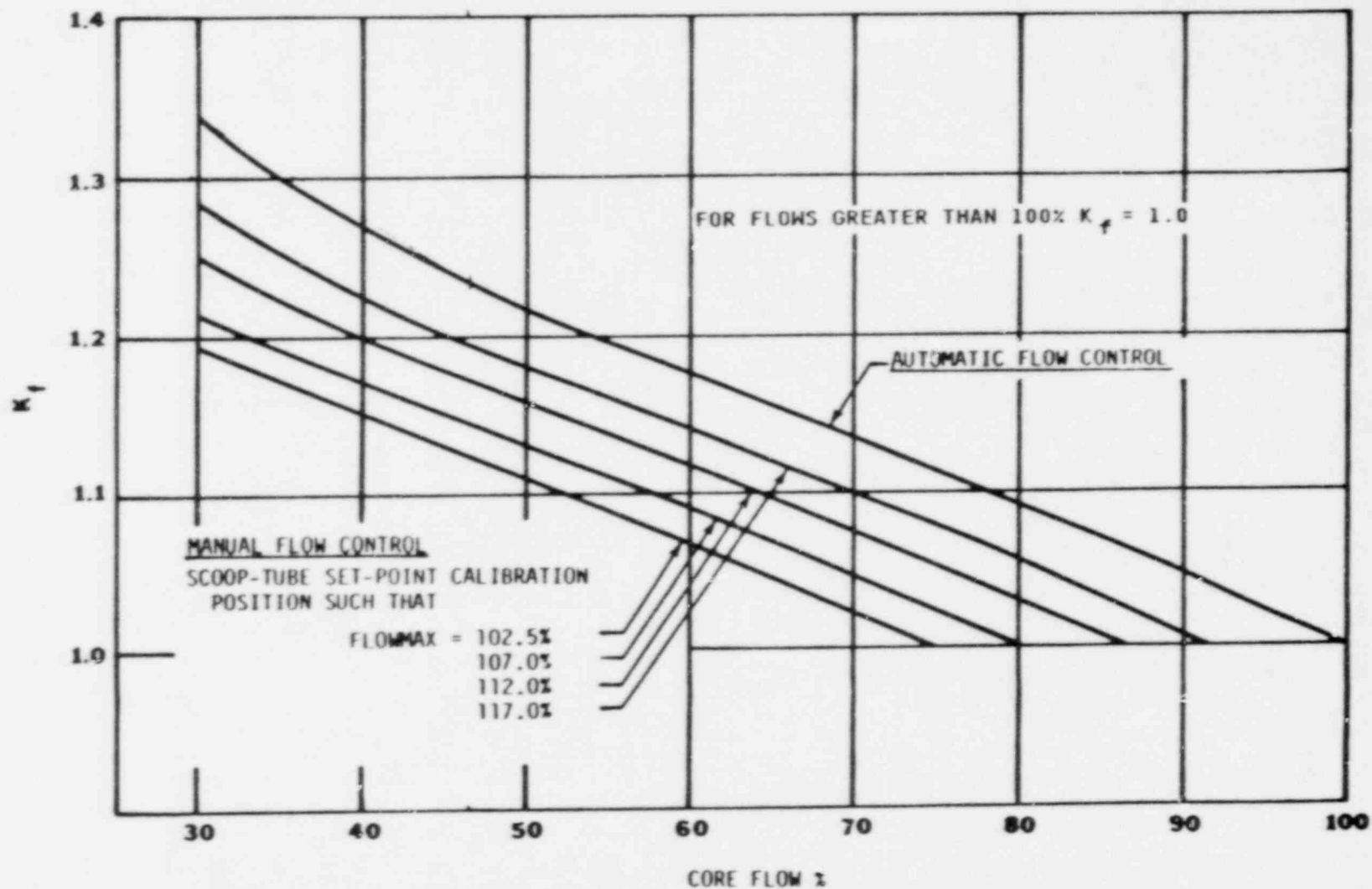


FIGURE 3.5-2

K_f FACTOR

QUAD CITIES
DPR-30

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;*
 $S \leq (.58WD + 58.5) \text{ FRP/MFLPD}$

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

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

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.

QUAD-CITIES
DPR-30

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.

GENERAL ELECTRIC COMPANY

AFFIDAVIT

I, Rudolph Villa, being duly sworn, depose and state as follows:

1. I am Manager, Consulting Services, General Electric Company, and have been delegated the function of reviewing the information described in paragraph 2 which is sought to be withheld and have been authorized to apply for its withholding.
2. The information sought to be withheld is contained in "Quad Cities Nuclear Power Station Units 1 & 2, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," NEDC-31345P, Revision 1, dated January 1988.
3. In designating material as proprietary, General Electric utilizes the definition of proprietary information and trade secrets set forth in the American Law Institute's Restatement of Torts, Section 757. This definition provides:

"A trade secret may consist of any formula, pattern, device or compilation of information which is used in one's business and which gives him an opportunity to obtain an advantage over competitors who do not know or use it.... A substantial element of secrecy must exist, so that, except by the use of improper means, there would be difficulty in acquiring information.... Some factors to be considered in determining whether given information is one's trade secret are: (1) the extent to which the information is known outside of his business; (2) the extent to which it is known by employees and others involved in his business; (3) the extent of measures taken by him to guard the secrecy of the information; (4) the value of the information to him and to his competitors; (5) the amount of effort or money expended by him in developing the information; (6) the ease or difficulty with which the information could be properly acquired or duplicated by others."

4. Some examples of categories of information which fit into the definition of proprietary information are:
 - a. Information that discloses a process, method or apparatus where prevention of its use by General Electric's competitors without license from General Electric constitutes a competitive economic advantage over other companies;

- b. Information consisting of supporting data and analyses, including test data, relative to a process, method or apparatus, the application of which provide a competitive economic advantage, e.g., by optimization or improved marketability;
 - c. Information which if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality or licensing of a similar product;
 - d. Information which reveals cost or price information, production capacities, budget levels or commercial strategies of General Electric, its customers or suppliers;
 - e. Information which reveals aspects of past, present or future General Electric customer-funded development plans and programs of potential commercial value to General Electric;
 - f. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection;
 - g. Information which General Electric must treat as proprietary according to agreements with other parties.
5. Initial approval of proprietary treatment of a document is typically made by the Subsection manager of the originating component, the person who is most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within the Company is limited on a "need to know" basis and such documents are clearly identified as proprietary.
6. The procedure for approval of external release of such a document typically requires review by the Subsection Manager, Project Manager, Principal Scientist or other equivalent authority, by the Subsection Manager of the cognizant Marketing function (or delegate) and by the Legal Operation for technical content, competitive effect and determination of the accuracy of the proprietary designation in accordance with the standards enumerated above. Disclosures outside General Electric are generally limited to regulatory bodies, customers and potential customers and their agents, suppliers and licensees then only with appropriate protection by applicable regulatory provisions or proprietary agreements.
7. The document mentioned in paragraph 2 above has been evaluated in accordance with the above criteria and procedures and has been found to contain information which is proprietary and which is customarily held in confidence by General Electric.

8. The information to the best of my knowledge and belief has consistently been held in confidence by the General Electric Company, no public disclosure has been made, and it is not available in public sources. All disclosures to third parties have been made pursuant to regulatory provisions of proprietary agreements which provide for maintenance of the information in confidence.
9. Public disclosure of the information sought to be withheld is likely to cause substantial harm to the competitive position of the General Electric Company and deprive or reduce the availability of profit making opportunities because it would provide other parties, including competitors, with valuable information on input parameters and analysis results of the SAFER/GESTR-LOCA analysis methodology, as well as details of current fuel designs that are not available to other parties.

STATE OF CALIFORNIA)
COUNTY OF SANTA CLARA) ss:

Rudolph Villa, being duly sworn, deposes and says:

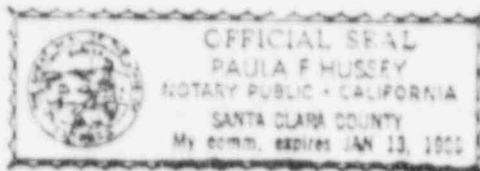
That he has read the foregoing affidavit and the matters stated therein are true and correct to the best of his knowledge, information, and belief.

Executed at San Jose, California, this 12th day of February, 1988.

Rudolph Villa

Rudolph Villa
General Electric Company

Subscribed and sworn before me this 12th day of February 1988.



Paula F. Hussey
NOTARY PUBLIC, STATE OF CALIFORNIA