May 30, 1985

Docket No. 50-265

Mr. Dennis L. Farrar
Director of Nuclear Licensing
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Farrar:

The Commission has issued the enclosed Amendment No. 86 to Facility Operating License No. DPR-30 for the Quad Cities Nuclear Power Station, Unit 2. The amendment consists of changes to the Technical Specifications (TS) in response to your applications dated January 3 and February 4, 1985.

The amendment revises the Technical Specifications to (1) incorporate new maximum average planar linear heat generation rate (MAPLHGR) curves for two new barrier fuel types to be used in the upcoming operating Cycle 8 and approve extended MAPLHGR curves for assembly average burnup of 45,000 MWD/ST for certain fuel types that will comprise part of the core for the upcoming operating Cycle 8; (2) change the calibration and functional test frequencies for certain specific instrumentation that is being modified into analog trip systems; and (3) incorporate appropriate Technical Specifications for operation with the newly modified scram discharge system.

A copy of the Safety Evaluation is also enclosed.

Sincerely,

Original signed by/

Roby B. Bevan, Project Manager
Operating Reactors Branch #2
Division of Licensing

Enclosures:
1. Amendment No. 86 to License No. DPR-30
2. Safety Evaluation

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Quad Cities Nuclear Power Station, Units 1 and 2

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

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

DOCKET NO. 50-265

QUAD CITIES NUCLEAR POWER STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 86
License No. DPR-30

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

   A. The applications for amendment by Commonwealth Edison Company (the licensee) dated January 3 and February 4, 1985, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-30 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 86 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Domenic B. Vassallo, Chief
Operating Reactors Branch #2
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 30, 1985

ATTACHMENT TO LICENSE AMENDMENT NO. 86

FACILITY OPERATING LICENSE NO. DPR-30

DOCKET NO. 50-265

Revise the Technical Specifications by deleting the following pages and inserting the enclosed pages.

Remove	Insert
3.1/4.1-2	3.1/4.1-2
-	3.1/4.1-2a
3.1/4.1-3	3.1/4.1-3
3.1/4.1-6	3.1/4.1-6
3.1/4.1-7	3.1/4.1-7
3.1/4.1-8	3.1/4.1-8
3.1/4.1-9	3.1/4.1-9
3.1/4.1-10	3.1/4.1-10
3.1/4.1-12	3.1/4.1-12
3.1/4.1-13	3.1/4.1-13
3.1/4.1-14	3.1/4.1-14
3.2/4.2-10	3.2/4.2-10
-	3.2/4.2-10a
3.2/4.2-14	3.2/4.2-14
-	3.2/4.2-14a
3.2/4.2-16	3.2/4.2-16
3.2/4.2-17	3.2/4.2-17
-	3.2/4.2-17a
3.3/4.3-3	3.3/4.3-3
Fig. 3.5-1 (5 pages)	Fig. 3.5-1 (6 pages)

3.1 LIMITING CONDITIONS FOR OPERATION BASES

The reactor protection system automatically initiates a reactor scram to:

a. preserve the integrity of the fuel cladding.

b. preserve the integrity of the primary system, and

c. minimize the energy which must be adsorbed and prevent criticality following a loss-of-coolant accident.

This specification provides the limiting conditions for operation necessary to preserve the ability of the system to tolerate single failures and still perform its intended function even during periods when instrument channels may be out of service because of maintenance. When necessary, one channel may be made inoperable for brief intervals to conduct required functional tests and calibrations.

The reactor protection system is of the dual channel type (reference SAR, Section 7.7.1.2). The system is made up of two independent trip systems, each having two subchannels of tripping devices. Each subchannel has an input from at least one instrument channel which monitors a critical parameter.

The outputs of the subchannels are combined in a one-out-of-two-logic; i.e., an input signal on either one or both of the subchannels will cause a trip system trip. The outputs of the trip systems are arranged so that a trip on both systems is required to produce a reactor scram.

This system meets the requirements of the IEEE 279 Standard for Nuclear Power Plant Protection Systems issued September 13, 1966. The system has a reliability greater than that of a two-out-of-three system and somewhat less than that of a one-out-of-two system (reference APED 5179).

With the exception of the average power range monitor (APRM) and intermediate range monitor (IRM) channels, each subchannel has one instrument channel. When the minimum condition for operation on the number of operable instrument channels per untripped protection trip system is met, or if it cannot be met and the affected protection trip system is placed in a tripped condition, the effectiveness of the protection system is preserved, i.e. the system can tolerate a single failure and still perform its intended function of scrammming the reactor. Three APRM instrument channels are provided for each protection trip system.

APRM's #1 and #3 operate contacts in one subchannel, and APRM's #2 and #3 operate contacts in the other subchannel. APRM's #4 and #5, and #6 are arranged similarly in the other protection trip system. Each protection trip system has one more APRM than is necessary to meet the minimum number required per channel. This allows the bypassing of one APRM per protection trip system for maintenance, testing, or calibration. Additional IRM channels have also been provided to allow for bypassing of one such channel. The bases for the scram settings for the IRM, APRM, high reactor pressure, reactor low water level, turbine control valve fast closure, and turbine stop valve closure are discussed in Specifications 2.1 and 2.2.

Pressure sensing of the drywell is provided to detect a loss-of-coolant accident and initiate the emergency core cooling equipment. The pressure-sensing instrumentation is a backup to the water-level instrumentation which is discussed in Specification 2.1. A scram is provided at the same setting as the emergency core cooling system (ECCS) initiation to minimize the energy which must be accommodated during a loss-of-coolant accident and to prevent the reactor from going critical following the accident.

The control rod drive scram system is designed so that all of the water which is discharged from the Reactor by a scram can be accommodated in the discharge piping. A part of this system is an individual instrument volume for each of the south and north CRD accumulators. These two volumes and their piping can hold in excess of 90 gallons of water and is the low point in the piping. No credit was taken for these volumes in the design of the discharge piping relative to the amount of water which must be accommodated during a scram. During normal operations, the discharge volumes are empty; however, should either volume fill with water, the water discharged to the piping from the Reactor may not be accommodated which could result in slow scram times or partial or no control rod insertion. To preclude this occurrence, level switches have been installed in both volumes which will alarm and scram the Reactor when the volume remaining in either instrument volume is approximately 40 gallons. For diversity of level sensing methods that will ensure and provide a scram, both differential pressure switches and thermal switches have been incorporated into the design and logic of the system. The setpoint for the scram signal has been chosen on the basis of providing sufficient volume remaining to accommodate a scram even with 5 gpm leakage per drive into SDV. As indicated above, there is sufficient volume in the piping to accommodate the scram without impairment of the scram times or the amount of insertion of the control rods. This function shuts the Reactor down while sufficient volume remains to accommodate the discharged water and precludes the situation in which a scram would be required but not be able to perform its function properly.

Loss of condensate vacuum occurs when the condenser can no longer handle heat input. Loss of condenser vacuum initiates a closure of the turbine stop valves and turbine bypass valves, which eliminates the heat input to the condenser. Closure of the turbine stop and bypass valves causes a pressure transient, neutron flux rise, and an increase in surface heat flux. To prevent the cladding safety limit from being exceeded if this occurs, a reactor scram occurs on turbine stop valve closure. The turbine stop valve closure scram function alone is adequate to prevent the cladding safety limit from being exceeded in the event of a turbine trip transient with bypass closure.

The condenser low-vacuum scram is a backup to the stop valve closure scram and causes a scram before the stop valves are closed, thus the resulting transient is less severe. Scram occurs at 21 inches Hg vacuum, stop valve closure occurs at 20 inches Hg vacuum, and bypass closure at 7 inches Hg vacuum.

High radiation levels in the main steamline tunnel above that due to the normal nitrogen and oxygen radioactivity are an indication of leaking fuel. A scram is initiated whenever such radiation level exceeds seven times normal background. The purpose of this scram is to reduce the source of such radiation to the extent necessary to prevent excessive turbine contamination. Discharge of excessive amounts of radioactivity to the site environs is prevented by the air ejector off-gas monitors, which cause an isolation of the main condenser off-gas line provided the limit specified in Specification 3.8 is exceeded.

The main steamline isolation valve closure scram is set to scram when the isolation valves are 10% closed from full open. This scram anticipates the pressure and flux transient which would occur when the valves close. By scramming at this setting, the resultant transient is insignificant.

A reactor mode switch is provided which actuates or bypasses the various scram functions appropriate to the particular plant operating status (reference SAR Section 7.7.1.2). Whenever the reactor mode switch is in the Refuel or Startup/Hot Standby position, the turbine condenser low-vacuum scram and main steamline isolation valve closure scram are bypassed. This bypass has been provided for flexibility during startup and to allow repairs to be made to the turbine condenser. While this bypass is in effect, protection is provided against pressure or flux increases by the high-pressure scram and APRM 15% scram, respectively, which are effective in this mode.

If the reactor were brought to a hot standby condition for repairs to the turbine condenser, the main steamline isolation valves would be closed. No hypothesized single failure or single operator action in this mode of operation can result in an unreviewed radiological release.

The manual scram function is active in all modes, thus providing for a manual means of rapidly inserting control rods during all modes of reactor operation.

The IRM system provides protection against excessive power levels and short reactor periods in the startup and intermediate power ranges (reference SAR Section 7.4.4.2 and 7.4.4.3). A source range monitor (SRM) system is also provided to supply additional neutron level information during startup but has no scram functions (reference SAR Section 7.4.3.2). Thus the IRM is required in the Refuel and Startup/Hot Standby modes. In addition, protection is provided in this range by the APRM 15% scram as discussed in the bases for Specification 2.1. In the power range the APRM system provides required protection (reference SAR Section 7.4.5.2). Thus, the IRM system is not required in the Run mode, the APRM's cover only the intermediate and power range, the IRM's provide adequate coverage in the startup and intermediate range.

The high-reactor pressure, high-drywell pressure, reactor low water level, and scram discharge volume high level scrams are required for the Startup/Hot Standby and Run modes of plant operation. They are therefore required to be operational for these modes of reactor operation.

The turbine condenser low vacuum scram is required only during power operation and must be bypassed to start up the unit.

to an out-of-limits input. This type of failure for analog devices is a rare occurrence and is detectable by an operator who observes that on signal does not track the other three. For purposes of analysis, it is assumed that this rare failure will be detected within 2 hours.

The bistable trip circuit which is a part of the Group 2 devices can sustain unsafe failures which are revealed only on test. Therefore, it is necessary to test them periodically.

A study was conducted of the instrumentation channels included in the Group 2 devices to calculate their 'unsafe' failure rates. The analog devices (sensors and amplifiers) are predicated to have an unsafe failure rate of less than 20 X 10^{-6} failures/hour. The bistable trip circuits are predicated to have an unsafe failure rate of less than 2 X 10^{-6} failures/hour. Considering the 2-hour monitoring interval for the analog devices as assumed above and a weekly test interval for the bistable trip circuits, the design reliability goal of 0.99999 is attained wity ample margin.

The bistable devices are monitored during plant operation to record their failure history and establish a test interval using the curve of Figure 4.1-1. There are numerous identical bistable devices used throughout the plant instrumentation system. Therefore, significant data on the failure rates for the bistable devices should be accumulated rapidly.

The frequency of calibration of the APRM flow biasing network has been established at each refueling outage. The flow biasing network is functionally tested at least once per month and, in addition, cross calibration checks of flow input to the flow-biasing network can be made during the functional test by direct meter reading (IEEE 279 Standard for Nuclear Power Plant Protection Systems, Section 4.9, September 13, 1966). There are several instruments which must be calibrated, and it will take several days to perform the calibration of the entire network. While the calibration is being performed, a zero flow signal will be sent to half of the APRM's, resulting in a half scram and rod block condition. Thus, if the calibration were performed during operation, flux shaping would not be possible. Based on experience at other generating stations, drift of instrument such as those in the flow biasing network is not significant; therefore, to avoid spurious scrams, a calibration frequency of each refueling outage is established.

Reactor low water level instruments 2-263-57A, 2-263-57B, 2-263-58A, and 2-263-58B have been modified to be an analog trip system. The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration and functional testing for instrument loops of the analog trip system, including reactor low water level, has been established in Licensing Topical Report NEDO-21617-A (December 1978). With the one-out-of-two-taken-twice logic, NEDO-21617-A states that each trip unit be subjected to a calibration/functional test frequency of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

Group 3 devices are active only during a given portion of the operational cycle. For example, the IRM is active during startup and inactive during full-power operation. Thus, the only test that is meaningful is the one performed just prior to shutdown or startup, i.e., the tests that are performed just prior to use of the instrument.

Calibration frequency of the instrument channel is divided into two groups. These are as follows:

1. passive type indicating devices that can be compared with like units on a continuous basis, and

2. vacuum tube or semiconductor devices and detectors that drift or lose sensitivity.

Experience with passive type instruments in Commonwealth Edison generating stations and substations indicates that the specified calibrations are adequate. For those devices which employ amplifiers, etc. drift specifications call for drift to be less than 0.4%/month i.e., in the period of a month a drift of 0.4% would occur, thus providing for adequate margin.

The sensitivity of LPRM detectors decreases with exposure to neutron flux at a slow and approximately constant rate. Changes in power distribution and electronic drift also require compensation. This compensation is accomplished by calibrating the APRM system every 7 days using heat balance data and by calibrating individual LPRM's at least every 1000 equivalent full-power hours using TIP traverse data. Calibration on this frequency assures plant operation at or below thermal limits.

A comparison of Tables 4.1-1 and 4.1-2 indicates that some instrument channels have not been included in the latter table. These are mode switch in shutdown, manual scram, high water level in scram discharge volume, main steamline isolation valve closure, turbine control valve fast closure, and turbine stop valve closure. All of the devices or sensors associated with these scram functions are simple on-off switches, hence calibration is not applicable, i.e., the switch is either on or off. Further, these switches are mounted solidly to the device and have a very low probability of moving; e.g., the thermal switches in the scram discharge volume tank. Based on the above, no calibration is required for these instrument channels.

B. The MFLPD shall be checked once per day to determine if the APRM scram requires adjustment. This may normally be done by checking the LPRM readings, TIP traces, or process computer calculations. Only a small number of control rods are moved daily, thus the peaking factors are not expected to change significantly and a daily check of the MFLPD is adequate.

References

1. I. M. Jacobs, "Reliability of Engineered Safety Features as a Function of Testing Frequency", Nuclear Safety, Vol. 9, No. 4, pp. 310-312, July - August, 1968.

2. Licensing Topical Report NEDO-21617-A (December 1978).

TABLE 3.1-1

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS REFUEL MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode Switch in shutdown		A
1	Manual scram		A
	IRM		
3	High flux	$\leq$ 120/125 of full scale	A
3	Inoperative		
	APRM[3]		
2	High flux (15% scram)	Specification 2.1.A.2	A
2	Inoperative		A
2 (per bank)	High water level in scram discharge volume[4]	$\leq$ 40 gallons per bank	A
2	High-reactor pressure	$\leq$ 1060 psig	A
2	High-drywell pressure[5]	$\leq$ 2 psig	A
2	Reactor low water level	$\geq$ 8 inches[8]	A
2	Turbine condenser low vacuum[7]	$\geq$ 21 inches Hg vacuum	A
2	Main steamline high radiation[12]	$\leq$ 7 X normal full power background	A
4	Main steamline isolation valve closure[7]	$\leq$ 10% valve closure	A

TABLE 3.1-2

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS STARTUP/HOT STANDBY MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System[1]	Trip Function	Trip Level Setting	Action[2]
1	Mode Switch in shutdown		A
1	Manual scram		A
	IRM		
3	High flux	≤ 120/125 of full scale	A
3	Inoperative		A
	APRM[3]		
2	High flux (15% scram)	Specification 2.1.A.2	A
2	Inoperative		A
2	High-reactor pressure	≤ 1060 psig	A
2	High-drywell pressure[5]	≤ 2 psig	A
2	Reactor low water level	≥ 8 inches[8]	A
2 (per bank)	High water level in scram discharge volume[4]	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum[7]	≥ 21 inches Hg vacuum	A
2	Main steamline high radiation[12]	≤ 7 X normal full power background	A
4	Main steamline isolation valve closure[7]	≤ 10% valve closure	A

TABLE 3.1-3

REACTOR PROTECTION SYSTEM (SCRAM) INSTRUMENTATION REQUIREMENTS RUN MODE

Minimum Number of Operable or Tripped Instrument Channels per Trip System(1)	Trip Function	Trip Level Setting	Action(2)
1	Mode Switch in shutdown		A
1	Manual scram		A
	APRM(3)		
2	High flux (flow biased)	Specification 2.1.A.1	A or B
2	Inoperative		A or B
2	Downscale(11)	≥ 3/125 of full scale	A or B
2	High-reactor pressure	≤ 1060 psig	A
2	High-drywell pressure	≤ 2 psig	A
2	Reactor low water level	≥ 8 inches(8)	A
2 (per bank)	High-water level in scram discharge volume	≤ 40 gallons per bank	A
2	Turbine condenser low vacuum	≥ 21 inches Hg vacuum	A or C
2	Main steamline high radiation(12)	≤ 7 X normal full power background	A or C
4	Main steamline isolation valve closure(6)	≤ 10% valve closure	A or C
2	Turbine control valve fast closure(9)	≥ 40% turbine/generator load mismatch (10)	A or C
2	Turbine stop valve closure(9)	≤ 10% valve closure	A or C
2	Turbine EHC control fluid low pressure(9)	≥ 900 psig	A or C

TABLE 4.1-1
SCRAM INSTRUMENTATION AND LOGIC SYSTEMS FUNCTIONAL TESTS

MINIMUM FUNCTIONAL TEST FREQUENCIES FOR SAFETY
INSTRUMENTATION, LOGIC SYSTEMS, AND CONTROL CIRCUITS

Instrument Channel	Group(3)	Functional Test(7)	Minimum Frequency(4)
Mode switch in shutdown	A	Place mode switch in shutdown	Each refueling outage
Manual scram	A	Trip channel and alarm	Every 3 months
IRM			
High flux	C	Trip channel and alarm (5)	Before each startup and weekly during refueling(6)
Inoperative	C	Trip channel and alarm	Before each startup and weekly during refueling (6)
APRM			
High flux	B	Trip output relays(5)	Once each week
Inoperative	B	Trip output relays	Once each week
Downscale	B	Trip output relays(5)	Once each week
High flux 15%	C	Trip output relays(5)	Before each startup and weekly during refueling(6)
High reactor pressure	A	Trip channel and alarm	(1)
High drywell pressure	A	Trip channel and alarm	(1)
Reactor low water level(2)	B	(8)	(1)
High water level in scram(9) discharge volume (thermal and dp switches)	A	Trip channel and alarm	Every 3 months
Turbine condenser low vacuum	A	Trip channel and alarm	(1)
Main steamline high radiation(2)	B	Trip channel and alarm(5)	Once each week
Main steamline isolation valve closure	A	Trip channel and alarm	(1)
Turbine control valve fast closure	A	Trip channel and alarm	(1)
Turbine stop valve closure	A	Trip channel and alarm	(1)
Turbine EHC control fluid low pressure	A	Trip channel and alarm	(1)

TABLE 4.1-1 (Cont'd)

Notes:

1. Initially once per month until exposure hours (M as defined on Figure 4.1-1) are 2.0 X 10^5; thereafter, according to Figure 4.1-1 with an interval not less than 1 month nor more than 3 months. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of Quad-Cities Units 1 and 2.

2. An instrument check shall be performed on low reactor water level once per day and on high steamline radiation once per shift.

3. A description of the three groups is included in the bases of this specification.

4. Functional tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.

5. This instrumentation is exempted from the instrument functional test definition (1.0 Definition F). This instrument functional test will consist of injecting a simulated electrical signal into the measurement channels.

6. Frequency need not exceed weekly.

7. A functional test of the logic of each channel is performed as indicated. This coupled with placing the mode switch in shutdown each refueling outage constitutes a logic system functional test of the scram system.

8. A functional test of the master and slave trip units is required monthly. A calibration of the trip unit is to be performed concurrent with the functional testing.

9. Only the electronics portion of the thermal switches will be tested using an electronic calibrator during the three month interval test. A water column or equivalent will be used to test the dp switches.

TABLE 4.1-2
SCRAM INSTRUMENT CALIBRATION

MINIMUM CALIBRATION FREQUENCIES FOR REACTOR PROTECTION INSTRUMENT CHANNELS

Instrument Channel	Group(1)	Calibration Standard(5)	Minimum Frequency(2)
High flux IRM	C	Comparison to APRM after heat balance	Every controlled shutdown(4)
High flux APRM			
Output signal	B	Heat balance	Once every 7 days
Flow bias	B	Standard pressure and voltage source	Refueling outage
LPRM	B(6)	Using TIP system	Every 1000 equivalent full power hours
High reactor pressure	A	Standard pressure source	Every 3 months
High drywell pressure	A	Standard pressure source	Every 3 months
Reactor low water level	B	Water level	(7)
Turbine condenser low vacuum	A	Standard vacuum source	Every 3 months
Main steamline high radiation	B	Appropriate radiation source(3)	Refueling outage
Turbine EHC control fluid Low pressure	A	Pressure source	Every 3 months
High water level in scram discharge volume (dp only)	A	Water level	Refueling Outage

Notes:

1. A description of the three groups is included in the bases of this specification.
2. Calibration tests are not required when the systems are not required to be operable or are tripped. If tests are missed, they shall be performed prior to returning the systems to an operable status.
3. A current source provides an instrument channel alignment every 3 months.
4. Maximum calibration frequency need not exceed once per week.
5. Response time is not part of the routine instrument check and calibration but will be checked every refueling outage.
6. Does not provide scram function.
7. Trip units are calibrated monthly concurrently with functional testing. Transmitters are calibrated once per operating cycle.

Optimizing each channel independently may not truly optimize the system considering the overall rules of system operation. However, true system optimization is a complex problem. The optimums are broad, not sharp, and optimizing the individual channels is generally adequate for the system.

The formula given above minimizes the unavailability of a single channel which must be bypassed during testing. The minimization of the unavailability is illustrated by curve 1 of Figure 4.2-2, which assumes that a channel has a failure rate of 0.1 X 10^{-6}/hour and 0.5 hour is required to test it. The unavailability is a minimum at a test interval i, of 3.6 X 10^{3} hours.

If two similar channels are used in a one-out-of-two configuration, the test interval for minimum availability changes as a function of the rules for testing. The simplest case is to test each one independent of the other. In this case, there is assumed to be a finite probability that both may be bypassed at one time. This case is shown by curve 2. Note that the unavailability is lower, as expected for a redundant system, and the minimum occurs at the same test interval. Thus, if the two channels are tested independently, the equation above yields the test interval for minimum unavailability.

A more usual case is that the testing is not done independently. If both channels are bypassed and tested at the same time, the result is shown in curve 3. Note that the minimum occurs at about 40,000 hours, much longer than for Cases 1 and 2. Also, the minimum is not nearly as low as Case 2, which indicates that this method of testing does not take full advantage of the redundant channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following the second channel be bypassed, tested, and restored. This is shown by curve 4. Note that there is not true minimum. The curve does have a definite knee, and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. This is, if the test interval is 4 months, test one of the other channels every 2 months. This is shown in curve 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

a. A one-out-of-n system may be treated the same as a single channel in terms of choosing a test interval.

b. More than one channel should not be bypassed for testing at any one time.

Reactor water level instruments 2-263-73A & B, HPCI high steam flow instruments 2-2389A-D, and HPCI steam line low pressure instruments 2-2352 & 2353 have been modified to be analog trip systems. The analog trip system consists of an analog sensor (transmitter) and a master/slave trip unit setup which ultimately drives a trip relay. The frequency of calibration and functional testing for instrument loops of the analog trip system has been established in Licensing Topical Report NEDO-21617-A (December 1978). With the one-out-of-two-taken-twice logic, NEDO-21617-A states that each trip unit be subjected to a calibration/functional test frequency of one month. An adequate calibration/surveillance test interval for the transmitter is once per operating cycle.

The radiation monitors in the ventilation duct and on the refueling floor which initiate building isolation and standby gas treatment operation are arranged in two one-out-of-two logic systems. The bases given above for the rod blocks apply here also and were used to arrive at the functional testing frequency.

Bases on experience at Dresden Unit 1 with instruments of similar design, a testing interval of once every 3 months has been found to be adequate.

The automatic pressure relief instrumentation can be considered to be a one-out-of-two logic system, and the discussion above applies to it also.

The instrumentation which is required for the postaccident condition will be tested and calibrated at regularly scheduled intervals. The basis for the calibration and testing of this instrumentation is the same as was discussed above for the reactor protection system and the emergency core cooling systems.

References

1. B. Epstein and A. Shiff, "Improving Availability and Readiness of Field Equipment Through Periodic Inspection", UCRL-50451, Laurence Radiation Laboratory, p. 10, Equation (24), July 16, 1968.

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)	$\leq 0.65W_D + 42$ (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 center line
3	IRM detector not in Startup position(8)	$\geq$ 2 feet below core center line
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

NOTE:

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

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

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

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

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

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

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

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

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

TABLE 4.2-1

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS INSTRUMENTATION, ROD BLOCKS, AND ISOLATIONS(7)

Instrument Channel	Instrument Functional Test(2)	Calibration (2)	Instrument Check(2)
ECCS Instrumentation			
1. Reactor low-low water level	(1)	Once/3 months	Once/day
2. Drywell high pressure	(1)	Once/3 months	None
3. Reactor low pressure	(1)	Once/3 months	None
4. Containment spray interlock			
a. 2/3 core height	(1) (10)	(10)	None
b. Containment pressure	(1)	Once/3 months	None
5. Low-pressure core cooling pump discharge	(1)	Once/3 months	None
6. Undervoltage 4-KV essential	Refueling outage	Refueling outage	None
7. Degraded voltage 4-KV essential busses	Refueling outage (8)	Refueling outage	Once/month
Rod Blocks			
1. APRM downscale	(1) (3)	Once/3 months	None
2. APRM flow variable	(1) (3)	Refueling outage	None
3. IRM upscale	(5) (3)	(5) (3)	None
4. IRM downscale	(5) (3)	(5) (3)	None
5. RBM upscale	(1) (3)	Refueling outage	None
6. RBM downscale	(1) (3)	Once/3 months	None
7. SRM upscale	(5) (3)	(5) (3)	None
8. SRM detector not in startup position	(5) (3)	(6)	None
9. IRM detector not in startup position	(5)	(6)	None
10. SRM downscale	(5) (3)	(5) (3)	None
11. High water level in scram discharge volume (SDV)	Once/3 months	Not applicable	None
12. SDV high level trip bypassed	Refueling outage	Not applicable	None
Main Steamline Isolation			
1. Steam tunnel high temperature	Refueling outage	Refueling outage	None
2. Steamline high flow	(1)	Once/3 months	Once/day
3. Steamline low pressure	(1)	Once/3 months	None
4. Steamline high radiation	(1) (4)	Refueling outage	Once/day
5. Reactor low low water level	(1) (10)	(10)	Once/day
RCIC Isolation			
1. Steamline high flow	Once/3 months (9)	Once/3 months (9)	None
2. Turbine area high temperature	Refueling outage	Refueling outage	None
3. Low reactor pressure	Once/3 months	Once/3 months	None

TABLE 4.2-1 (Cont'd)

Instrument Channel	Instrument Functional Test (2)	Calibration (2)	Instrument Check (2)
HPCI Isolation			
1. Steamline high flow	(1) (9) (10)	(9) (10)	None
2. Steamline area high temperature	Refueling outage	Refueling outage	None
3. Low reactor pressure	(1) (10)	(10)	None
Reactor Building Ventilation System Isolation and SBGTS Initiation			
1. Refueling floor radiation monitors	(1)	Once/3 months	Once/day
Control Room Ventilation System Isolation			
1. Reactor low water level	(1)	Once/3 months	Once/day
2. Drywell high pressure	(1)	Once/3 months	None
3. Main steamline high flow	(1)	Once/3 months	Once/day

Notes:

1. Initially once per month until exposure hours (M as defined on Figure 4.1-1) are 2.0×10^5; thereafter, according to Figure 4.1-1 with an interval not less than 1 month nor more than 3 months. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of Quad-Cities Units 1 and 2.

2. Functional tests, calibrations, and instrument checks are not required when these instruments are not required to be operable, or are tripped.

3. This instrumentation is excepted from the functional test definition. The functional test shall consist of injecting a simulated electrical signal into the measurement channel.

4. This instrument channel is excepted from the functional test definitions and shall be calibrated using simulated electrical signals once every 3 months.

5. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed during each startup or during controlled shutdowns with a required frequency not to exceed once per week.

Notes: (Cont)

6. The positioning mechanism shall be calibrated every refueling outage.

7. Logic system functional tests are performed as specified in the applicable section for these systems.

8. Functional tests shall include verification of operation of the degraded voltage. 5 minute timer and 7 second inherent timer.

9. Verification of the time delay setting of $3 \leq \tau \leq 10$ seconds shall be performed during each refueling outage.

10. Trip units are functionally tested monthly. A calibration of the trip unit is to be performed concurrent with the functional testing. Transmitters are calibrated once per operating cycle.

3. The control rod drive housing support system shall be in place during reactor power operation and when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel unless all control rods are fully inserted and Specification 3.3.A.1 is met.

   a. Control rod withdrawal sequences shall be established so that maximum reactivity that could be added by dropout of any increment of any one control blade would be such that the rod drop accident design limit of 280 cal/gm. is not exceeded.

   b. Whenever the reactor is in the Startup/Hot Standby or Run mode below 20% rated thermal power, the rod worth minimizer shall be operable. A second operator or qualified technical person may be used as a substitute for an inoperable rod worth minimizer which fails after withdrawal of at least 12 control rods to the fully withdrawn position. The rod worth minimizer may also be bypassed for low power physics testing to demonstrate the shutdown margin requirements of Specification 3.3.A if a nuclear engineer is present and verifies the step-by-step rod movements of the test procedure.

4. Control rods shall not be withdrawn for startup or refueling unless at least two source range channels have an observed count rate equal to or greater than three counts per seconds and these SRM's are fully inserted.

5. During operation with limiting control rod patterns, as determined by the nuclear engineer, either:

   a. both RBM channels shall be operable.

   b. control rod withdrawal shall be blocked; or

3. The correctness of the control rod withdrawal sequence input to the RWM computer shall be verified after loading the sequence.

   Prior to the start of control rod withdrawal towards criticality, the capability of the rod worth minimizer to properly fulfill its function shall be verified by the following checks:

   a. The RWM computer on line diagnostic test shall be successfully performed.

   b. Proper annunciation of the selection error of one out-of-sequence control rod shall be verified.

   c. The rod block function of the RWM shall be verified by withdrawing the first rod as an out-of-sequence control rod no more than to the block point.

4. Prior to control rod withdrawal for startup or during refueling, verify that a least two source range channels have an observed count rate of at least three counts per second.

5. When a limiting control rod pattern exists, an instrument functional test of the RBM shall be performed prior to withdrawal of the designated rod(s) and daily thereafter.

6. The scram discharge volume vent and drain valves shall be verified open at least once per 31 days. Theses valves may be closed intermittently for testing under administrative control and at least once per 92 days, each valve shall be cycled through at least one complete cycle of full travel. At least once each Refueling Outage, the scram discharge volume vent and drain valves will be demonstrated to:

   a. Close within 30 seconds after receipt of a signal for control rods to scram, and

   b. Open when the scram signal is reset.

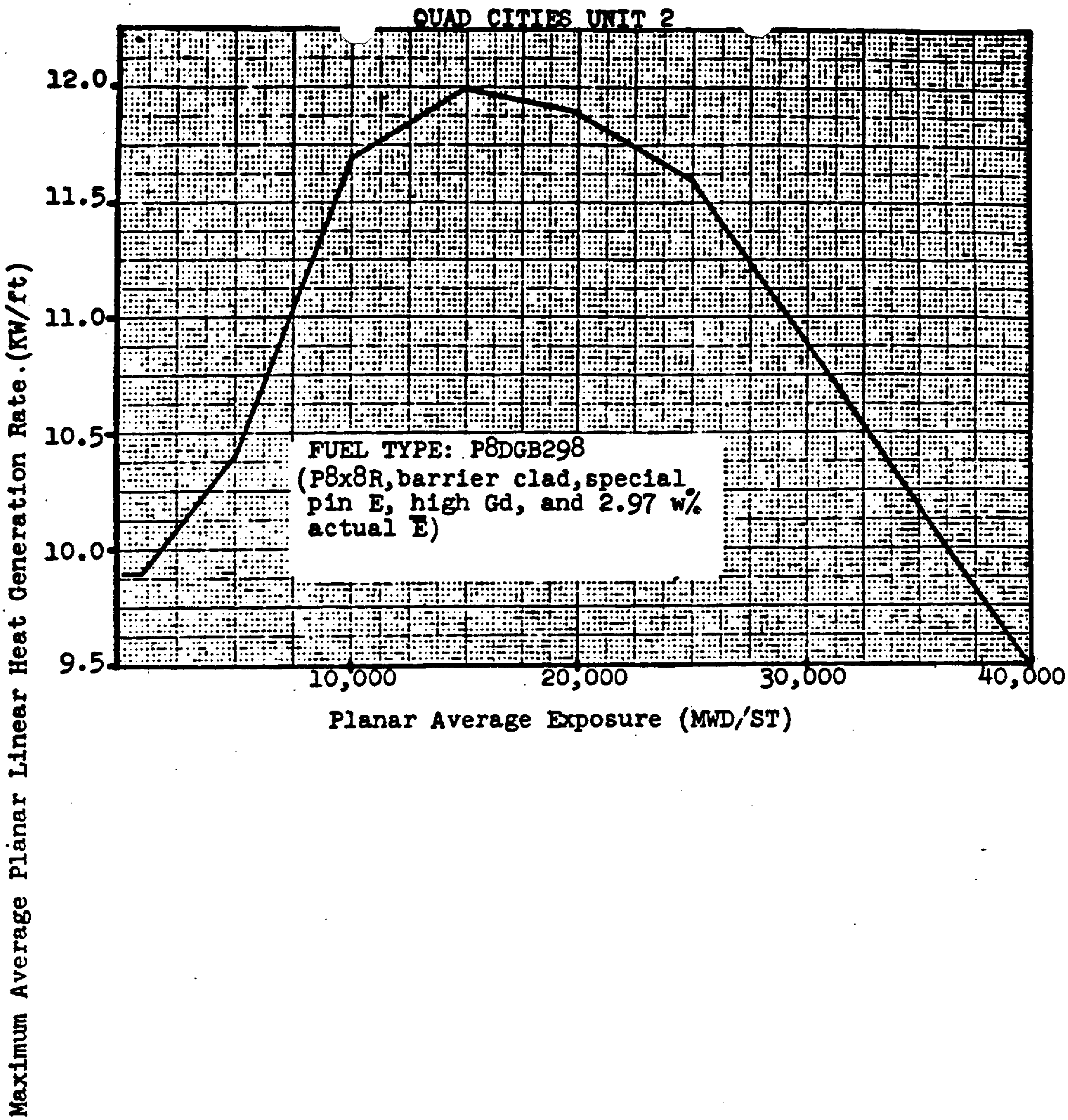

Figure 3.5-1
(Sheet 1 of 6)

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

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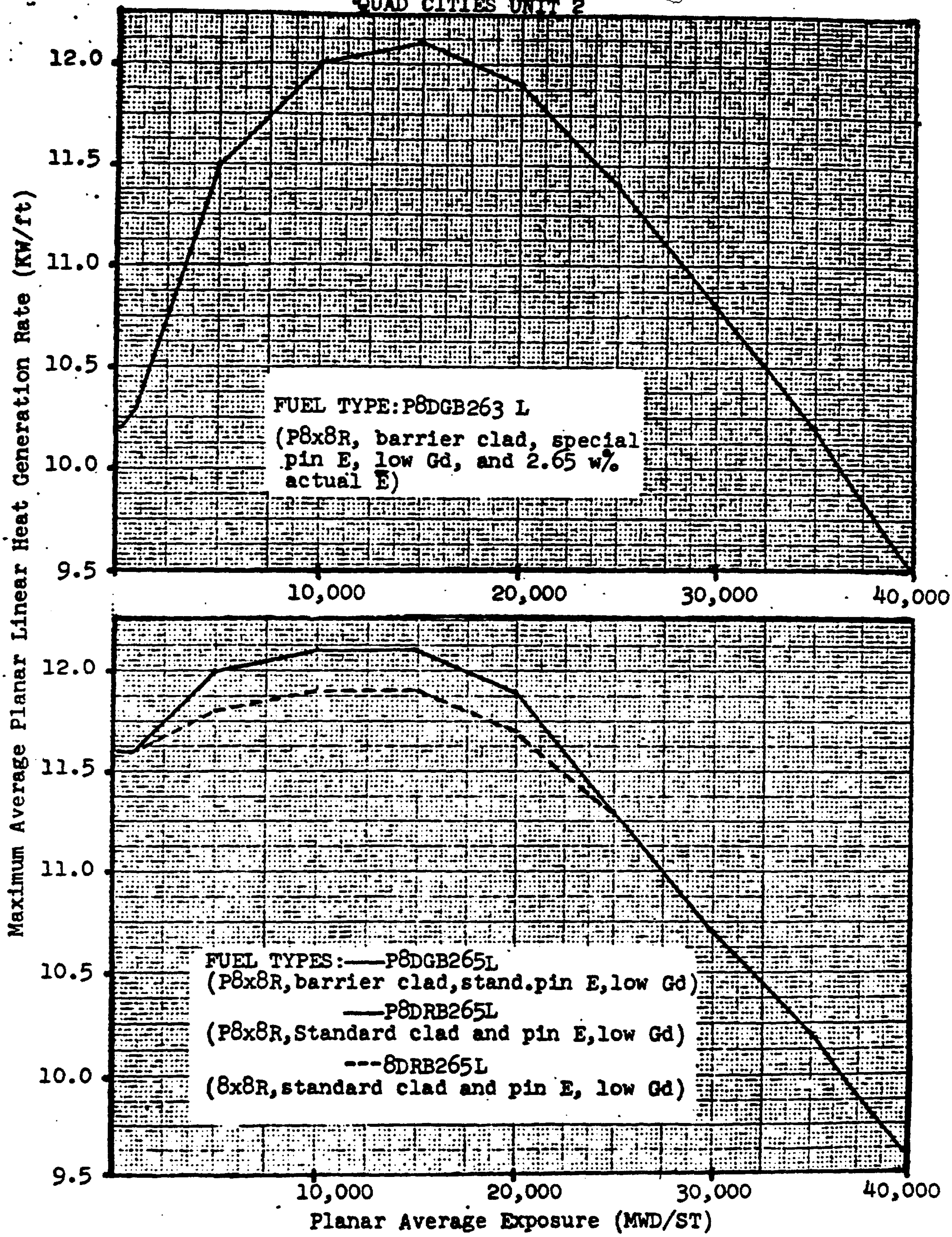


Figure 3.5-1 (Sheet 2 of 6) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

Quad Cities Unit 2

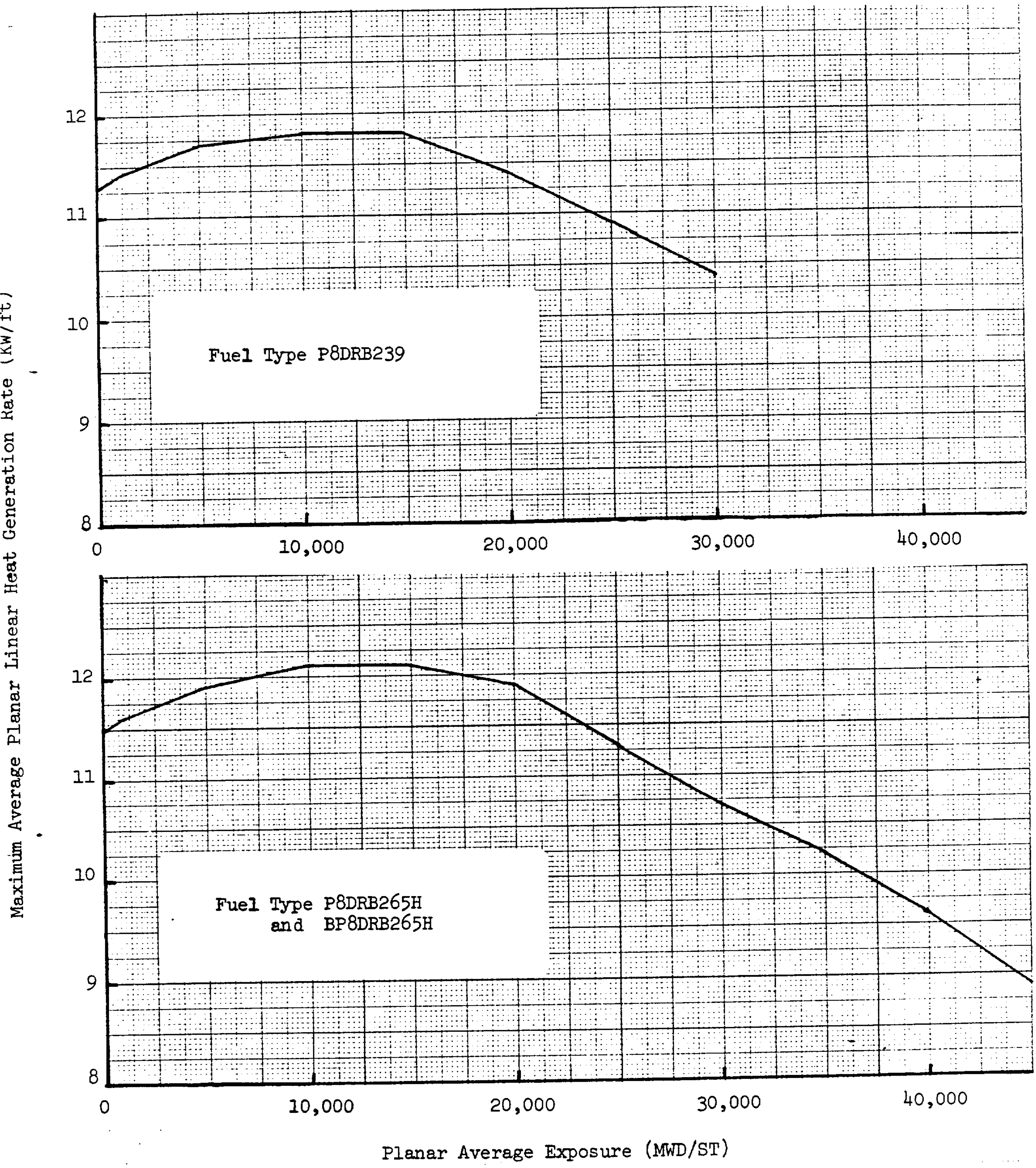
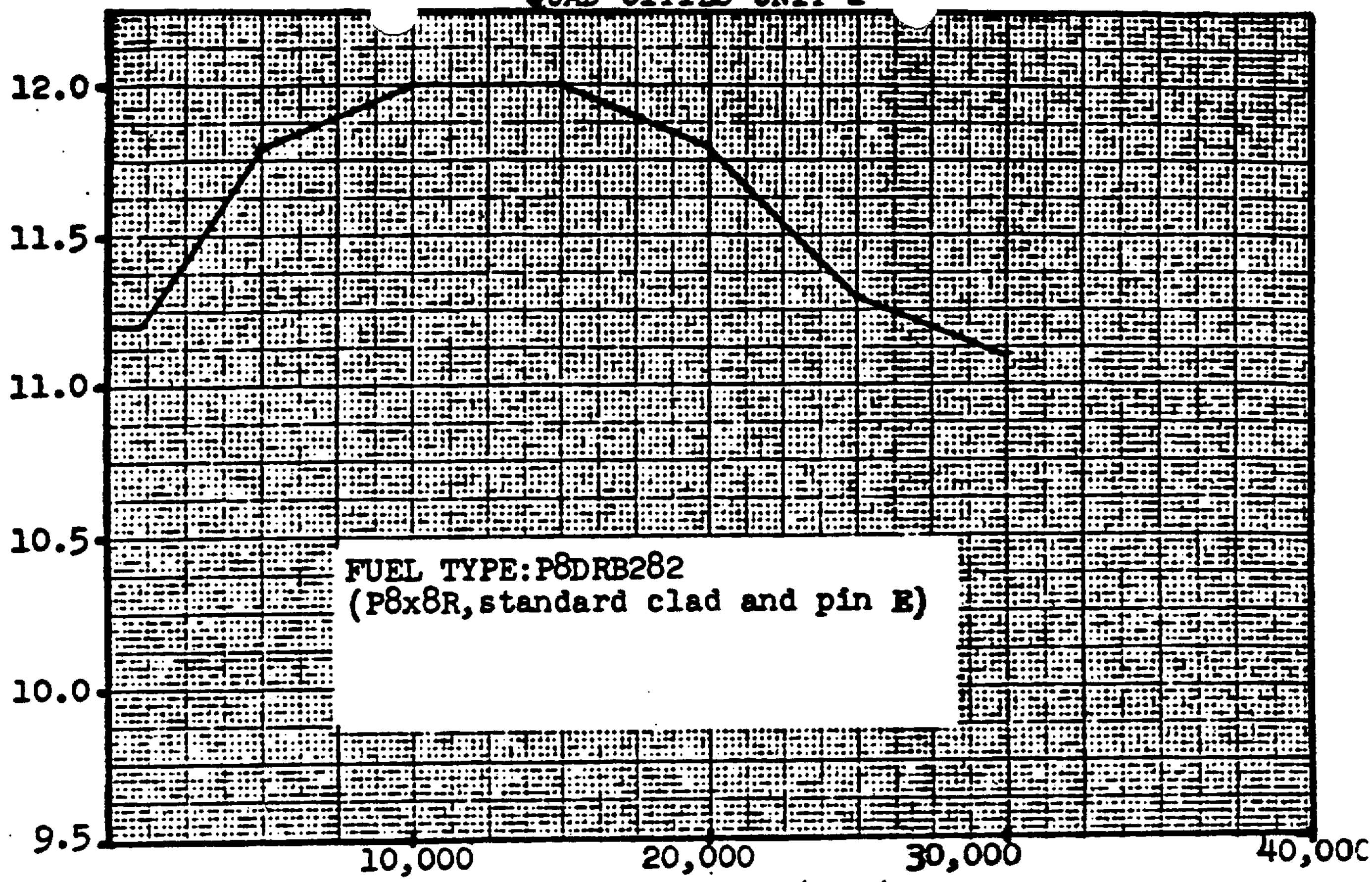
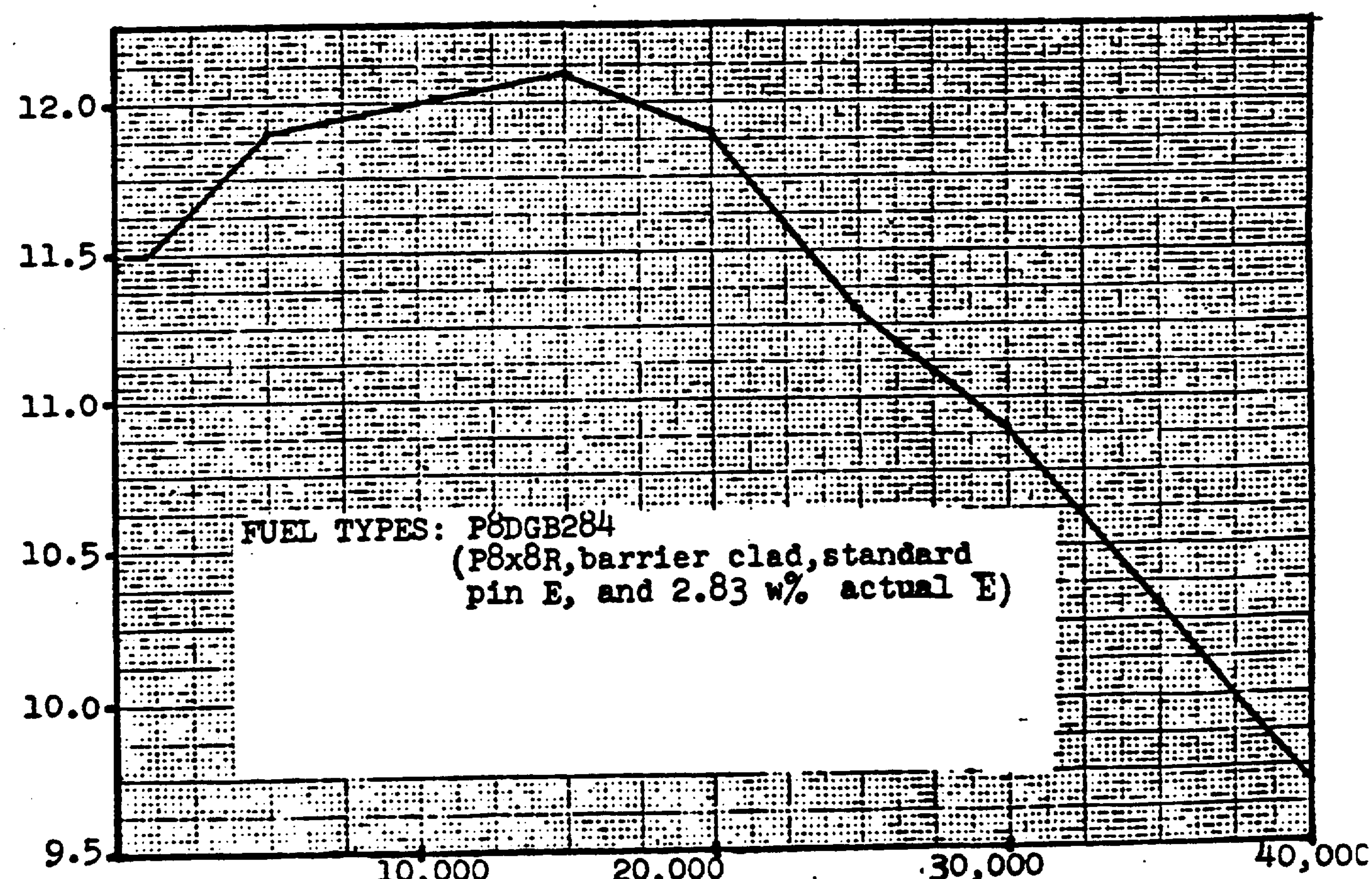


Figure 3.5-1 Maximum Average Planar Linear Heat Generation

(Sheet 3 of 6) Rate (MAPLHGR) vs. Planar Average Exposures





Maximum Average Planar Linear Heat Generation Rate (KW/ft)

Planar Average Exposure (MWD/ST)

Figure 3.5-1 (Sheet 4 of 6) Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) vs. Planar Average Exposure

Amendment No. ~~54~~, 69, 86

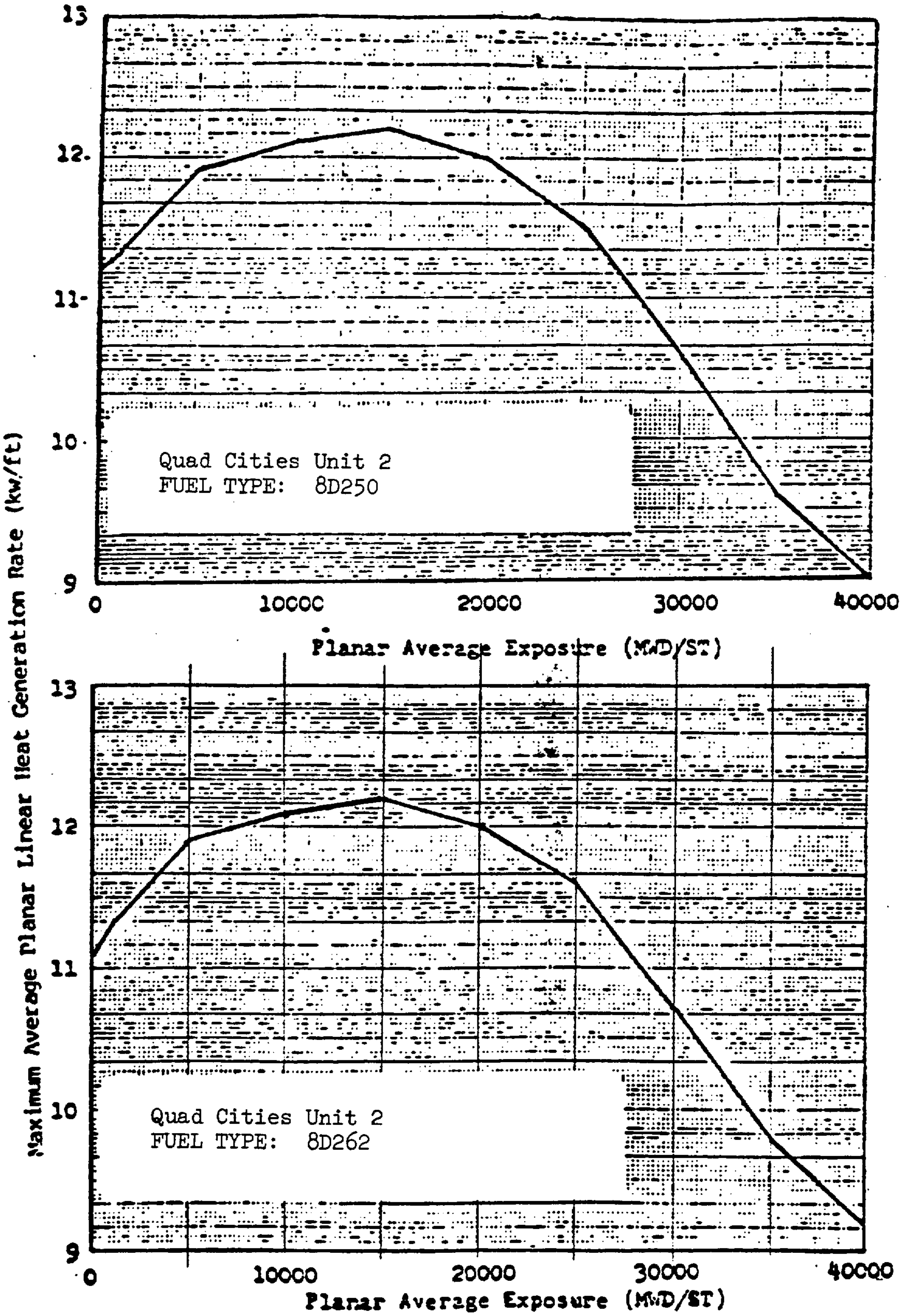
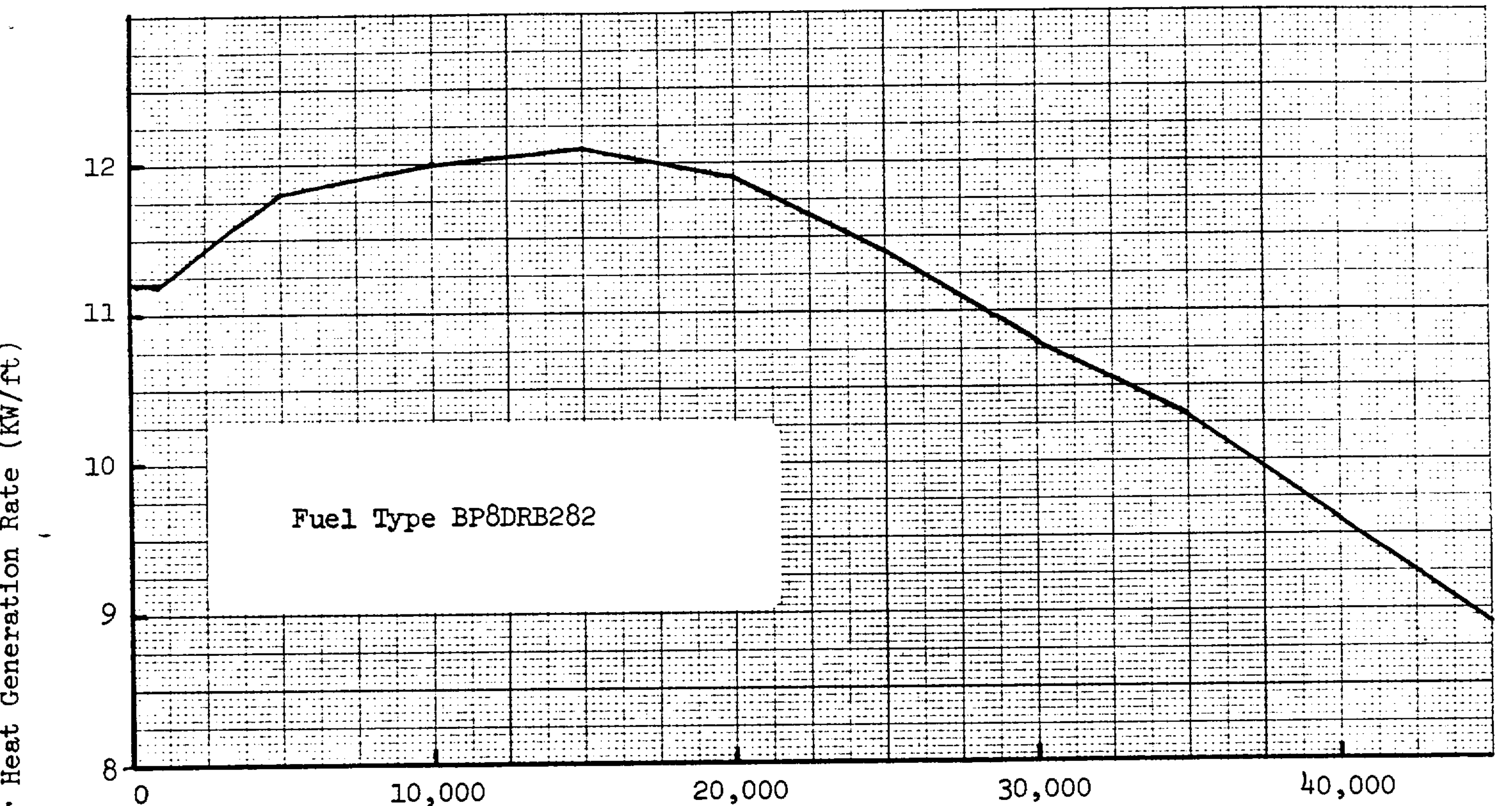


Figure 3.5-1 Maximum Average Planar Linear Heat
(Sheet 5 of 6) Generation Rate (MAPLHGR) vs. Planar
Average Exposure



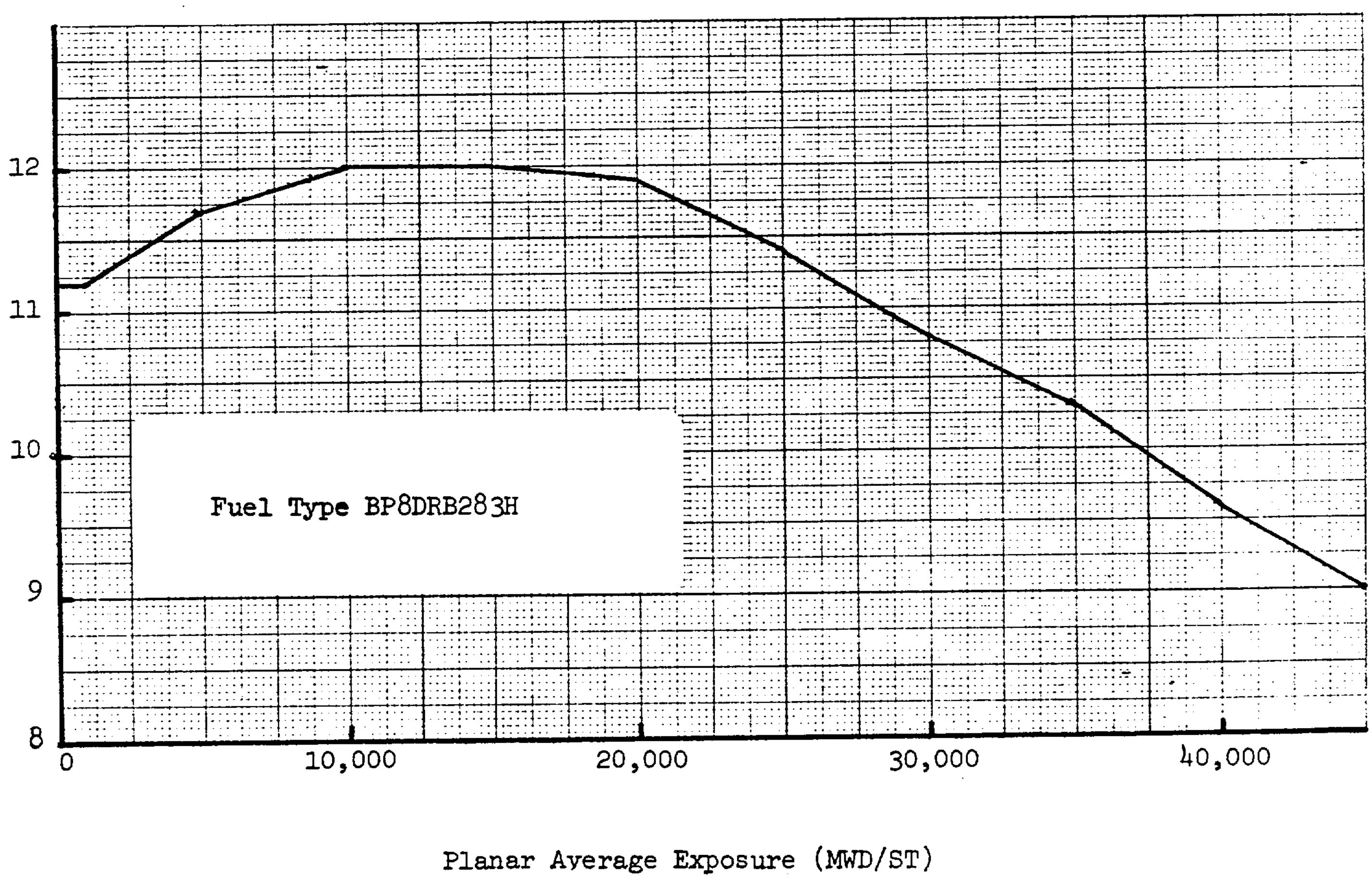


Figure 3.5-1 Maximum Average Planar Linear Heat Generation
(Sheet 6 of 6) Rate (MAPLHGR) vs. Planar Average Exposure

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 86 TO FACILITY OPERATING LICENSE NO. DPR-30

COMMONWEALTH EDISON COMPANY
AND
IOWA-ILLINOIS GAS AND ELECTRIC COMPANY

QUAD CITIES STATION, UNIT 2

DOCKET NO. 50-265

1.0 INTRODUCTION

By letters dated January 3 and February 4, 1985, Commonwealth Edison Company (CECo, the licensee) proposed changes to the Technical Specifications (TS) for Quad Cities Unit 2 (see References 1 and 2). These changes would (1) incorporate new maximum average planar linear heat generation rate (MAPLHGR) curves for two new barrier fuel types to be used in the upcoming operating Cycle 8, and approve MAPLHGR curves for assembly average burnup to 45,000 MWD/ST for certain fuel types contained in the core during upcoming operating Cycle 8; (2) change the calibration and functional test frequencies for certain specific instrumentation that is being replaced with analog trip systems; and (3) incorporate appropriate TS for operation with the newly modified scram discharge system.

2.0 EVALUATION

2.1 MAPLHGR Limits

The reference document containing the emergency core cooling system (ECCS) analysis for Quad Cities Unit 1 and 2 (as well as Dresden Units 2 and 3) has previously been approved by the NRC staff and continues to be the basis for MAPLHGR limits for all fuel types used in these units (Reference 4). This reference document has been updated as appropriate for other fuel types by issuance of Errata and Addenda (E and A) to Reference 4. Barrier fuel types BP8DRB282 and BP8DRB283H, to be used in the core during the upcoming operating Cycle 8 for Quad Cities Unit 2, have previously been approved for use in Quad Cities Unit 1 (Reference 3) and are currently being used in operating Cycle 8 of that nuclear unit. The licensee's application (Reference 2) provides E and A No. 13 to the previously approved ECCS analysis for Quad Cities Unit 1 and 2 (Reference 4). As did E and A No. 12 for Quad Cities Unit 1, so does E and A No. 13 to Reference 4 show that the MAPLHGR curves for fuel types BP8DRB282 and BP8DRB283H satisfy the requirements of the 10 CFR 50.46 acceptance criteria, and are appropriate for incorporation into the TS for Quad Cities Unit 2, so that the licensee's proposal to change the TS to incorporate those MAPLHGR curves into the TS is acceptable.

The licensee's application (Reference 2) also contains a request to approve an extension of MAPLHGR limits to assembly average burnup of 45,000 MWD/ST for two fuel types already in the core. By E and A No. 10 to Reference 4, provided in the licensee's application (Reference 2), the licensee supports an extension to MAPLHGR curves from 30,000 to 45,000 MWD/ST for fuel types P8DRB265H and BP8DRB265H. These limits were generated by method previously approved and having general applicability for MAPLHGR limit determination (Reference 5). In response to NRC staff concerns that the possible effects of enhanced fission gas release were not adequately considered in the fuel performance model, GE requested that credit for approved but unapplied ECCS evaluation model changes and calculated peak cladding temperature margin be used to avoid MAPLHGR penalties at high burnups (References 6 and 7). This proposal was found acceptable (Reference 8) provided that certain plant-specific conditions were met. As documented in the licensee's application (Reference 2), the General Electric Company (GE)-produced supporting reference document demonstrates the applicability to Quad Cities Unit 2 operating Cycle 8. Further, the staff has investigated potential changes in the radiological consequences of the fuel handling accident (the design basis accident) due to the possible involvement of extended burnup (as high as 45,000 MWd/MT) fuel in the accident scenario, and concludes that new radiological consequences would not be in excess of the Standard Review Plan (SRP) guideline value of 75 REM to the thyroid. On the basis of the foregoing considerations, the extended MAPLHGR limits for the fuel types cited are acceptable.

2.2 Analog Trip Instrumentation Surveillance Frequency

Certain equipment is being replaced to satisfy the requirement of 10 CFR 50.49 regarding environmental qualification of electrical equipment important to safety. In association with these changes, several existing instruments will be converted into analog trip systems; these are:

Reactor Low Water Level Instrument, 2-263-57A and B and 2-263-58A and B
Reactor Water High Level Instrument, 2-263-73A and B
High Pressure Coolant Injection (HPCI) High Steam Flow Instrument, 2-2389A through D
HPCI Steam Line Low Pressure Instrument, 2-2352 and 2-2353

The analog trip systems consist of an analog sensor and transmitter, and a trip unit arrangement which ultimately actuates a trip relay. The frequency of calibration and functional testing for instrument loops of the analog trip system has been established in Reference 9, an NRC-approved reference document. With the currently installed one-out-of-two, taken twice logic, the prescribed calibration/functional test frequency is once per month. The proper calibration/functional test frequency for the respective transmitters, however, is once per operating cycle. The TS changes proposed in Reference 1 are essentially identical to those previously approved for Unit 1 (Reference 3), and would require the channel calibration to be performed at the transmitter at a frequency of once per operating cycle. Since this is the calibration frequency recommended in the NRC-approved GE Topical Report, NEDO-21617-A (Reference 9), and the proposal is to conform the surveillance requirements to the recommended and NRC-approved period, the licensee's proposal is acceptable.

2.3 Scram Discharge System

A Generic Safety Evaluation for the modified scram discharge system, issued December 10, 1980, endorsed the criteria set forth by the BWR Owners Subgroup to meet the concerns arising from the Browns Ferry incomplete scram event of July 1980. By the NRC Confirmatory Order of June 24, 1982, the licensee's commitment to modify its scram discharge system in response to these concerns was confirmed (Reference 10). Also, model Technical Specifications were forwarded to the licensee as guidance for revising the TS for operation with the newly modified scram discharge system. Following a period of discussion with the licensee regarding the application of the model Technical Specifications to the unit-specific TS for Quad Cities, Units 1 and 2, the licensee by letter dated January 3, 1985 proposed TS for the newly modified scram discharge system for Unit 2. The TS changes proposed in the licensee's submittal are essentially identical to those previously approved for Unit 1 (Reference 3), are fully responsive to the concerns addressed in the Generic Safety Evaluation on scram discharge systems, are in keeping with the guidance provided in the model Technical Specifications, and are acceptable.

3.0 ENVIRONMENTAL CONSIDERATIONS

The amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

4.0 CONCLUSION

We have 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.

Principal Contributor: R. B. Bevan

Dated: May 30, 1985

REFERENCES

1) Letter, R. Rybak (CECo) to H. Denton (NRC), dated January 3, 1985.

2) Letter, R. Rybak (CECo) to H. Denton (NRC), dated February 4, 1985.

3) Letter. R. Bevan (NRC) to D. Farrar (CECo), dated August 2, 1984.

4) GE Topical Report, NEDO-24146-A, "Loss-of-Coolant Accident Analysis Report for Dresden Units 2,3 and Quad Cities Units 1,2 Nuclear Power Stations," Rev. 1, dated April 1979, as subsequently revised by Errata and Addenda 1 through 13.

5) Letter, D. Eisenhut (NRC) to E. Fuller (GE), dated May 30, 1977.

6) Letter, R. Engel (GE) to T. Ippolito (NRC), dated May 6, 1981.

7) Letter, R. Engel (GE) to T. Ippolito (NRC), dated May 28, 1981.

8) Memorandum, L. Rubenstein (NRC) for T. Novak (NRC), "Extension of GE-ECCS Performance Limits," dated June 25, 1981.

9) GE Topical Report, NEDO-21617-A, "Analog Transmitter/Trip Unit Systems for Engineered Safeguard Sensor Trip Inputs," dated December 1978.

10) Letter, D. Eisenhut (NRC) to D. Farrar (CECo), dated June 24, 1982.