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Chicago, Illinois 60690

REGULATORY DOCKET FILE COPY

September 13, 1978

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden Station Units 2 & 3
Quad-Cities Station Units 1 & 2
Proposed Amendment to Facility
Operating License Nos. DPR-19,
DPR-25, DPR-29 and DPR-30 Associated
with Reanalysis of the Loss-of-
Coolant Accident
NRC Docket Nos. 50-237/249/254/265

Reference (a): R. L. Bolger letter to E. G. Case
dated October 3, 1977

Dear Sir:

Pursuant to 10 CFR 50.59, Commonwealth Edison proposes to amend the Dresden Station Units 2, 3 and Quad-Cities Station Units 1, 2 Technical Specifications regarding revised Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) limits associated with reanalysis of the Loss-of-Coolant Accident. Reference (a) transmitted the General Electric Report NEDO-24046, dated August 1977, entitled "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)." NEDO-24046 documents the completion of the ECCS reevaluation and forms the basis for the revised MAPLHGR limits.

During September 1976, General Electric initiated an input reverification program to remove known conservatisms in the 1975 Appendix K ECCS Evaluations. By December 1976, the reverification program had identified significant input errors in the reflood calculations. Additionally, for the first four BWR/3 plant analysis performed in 1975, a double credit for structural absorption was discovered.

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On December 15, 1976, Commonwealth Edison derated Dresden 2, 3 and Quad-Cities 1, 2 limits (MAPLHGR) by 4% to eliminate the double credit for structural absorption. In January 1977, General Electric estimated the net effect of input errors to be a 6.5% decrease in MAPLHGRs. Commonwealth Edison implemented the estimated reduction, on an interim basis, pending a final ECCS reevaluation.

Report NEDO-24046 documents the completion of the required ECCS reevaluation. The analysis was performed using recent NRC approved model changes in the General Electric computer codes. These changes remove unnecessary conservatism and tend to counter the effects of the input errors.

Section 5 of the analysis report presents the input and model changes to the ECCS reevaluation. The input changes identified, with the exception of a more accurate DBA break size and peripheral bypass area, represent corrections to erroneous input values used in previous analysis. Generally, the input changes increase the delay to core reflood (more severe results).

Evaluation model changes, also presented in Section 5, are twofold. The first model change was required by NRC as a penalty in calculating the counter current flow limiting (CCFL) effect on additional delay in ultimate reflood. The second model change, in the assembly heatup calculation, removes conservatism in the radiation and conduction equations.

Table 6 presents the single-failure analysis. These results are unchanged from the previous analysis.

Attachment I contains the proposed changes for Dresden Units 2 & 3 and Attachment II contains the proposed changes for Quad-Cities Units 1 & 2. The referenced NEDO-24046 document and the attached Technical Specification changes have received onsite and offsite review and approval.

Pursuant to 10 CFR 170, Commonwealth Edison has determined that the proposed amendment is a combined Class III and Class I for each site. As such, we have enclosed a fee remittance in the amount of $8,800.00.

Three (3) signed originals and fifty-seven (57) copies of this transmittal are provided for your use.

Very truly yours,

C. Reed

Cordell Reed
Assistant Vice-President

attachment

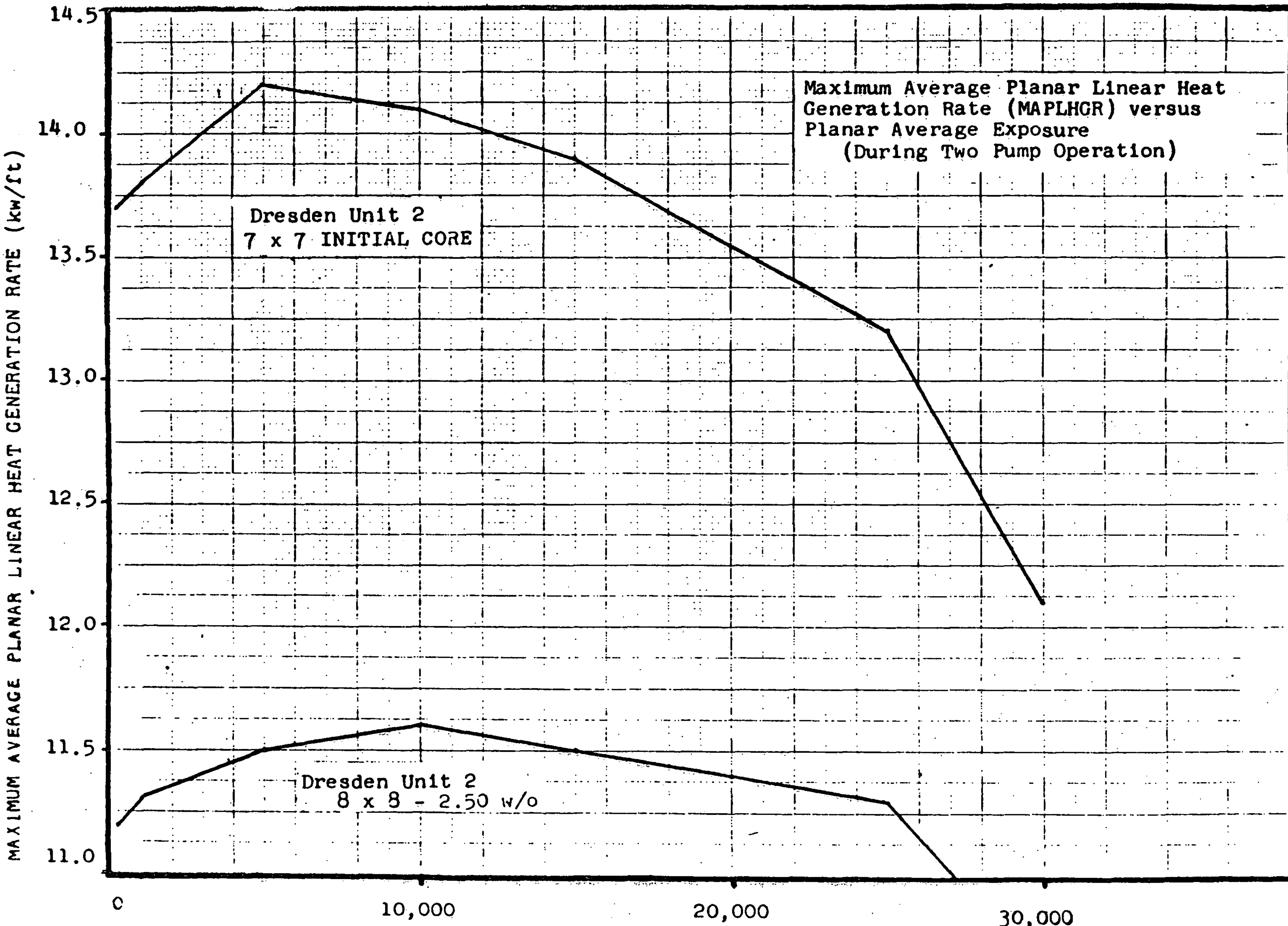
SUBSCRIBED and SWORN to
before me this 14th, day
of September, 1978.

Nancy M. Dascenzo
Notary Public

ATTACHMENT I

DRESDEN STATION UNITS 2 & 3

NRC DOCKET NOS. 50-237/249


Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure (During Two Pump Operation)
Dresden Unit 2
7 x 7 INITIAL CORE
Dresden Unit 2
8 x 8 - 2.50 w/o
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (kw/ft)
14.5
14.0
13.5
13.0
12.5
12.0
11.5
11.0
0
10,000
20,000
30,000
PLANAR AVERAGE EXPOSURE (MWD/T)
FIGURE 3.5.1.B

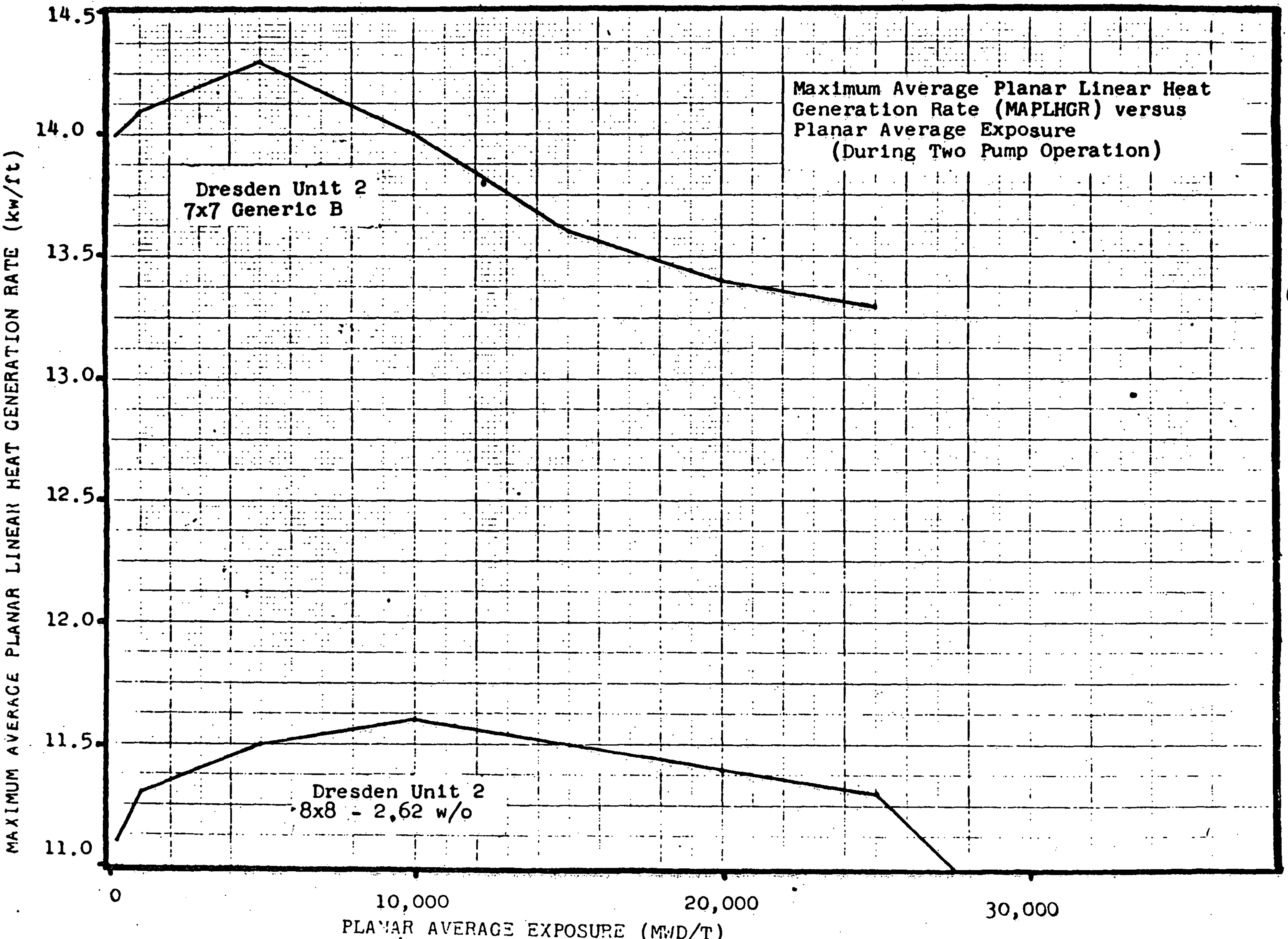

Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure (During Two Pump Operation)
Dresden Unit 2
7x7 Generic B
Dresden Unit 2
8x8 - 2.62 w/o
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (kw/ft)
14.5
14.0
13.5
13.0
12.5
12.0
11.5
11.0
0
10,000
20,000
30,000
PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.B

3.5 Limiting Conditions for Operation Bases

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

Based on the loss of coolant analyses included in References (1) and (2) in accordance with 10CFR50.46 and Appendix K, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident, to limit the calculated peak clad temperature 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%.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the

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(1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2,3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)", NEDO-24046, August 1977.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

(3) APED-"Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards" - April 1969, I.M. Jacobs and P.W. Marriott.

3.5 Limiting Condition for Operation Bases (Cont'd)

I. Average Planar LHGR

This specification assures that the peak cladding temperature following a postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10CFR50 Appendix K considering the postulated effects of fuel pellet densification.

The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1). Power operation with LHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit. Those values represent limits for operation to ensure conformance with 10CFR50 and Appendix K only if they are more limiting than other design parameters. The maximum average planar LHGRs plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs 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

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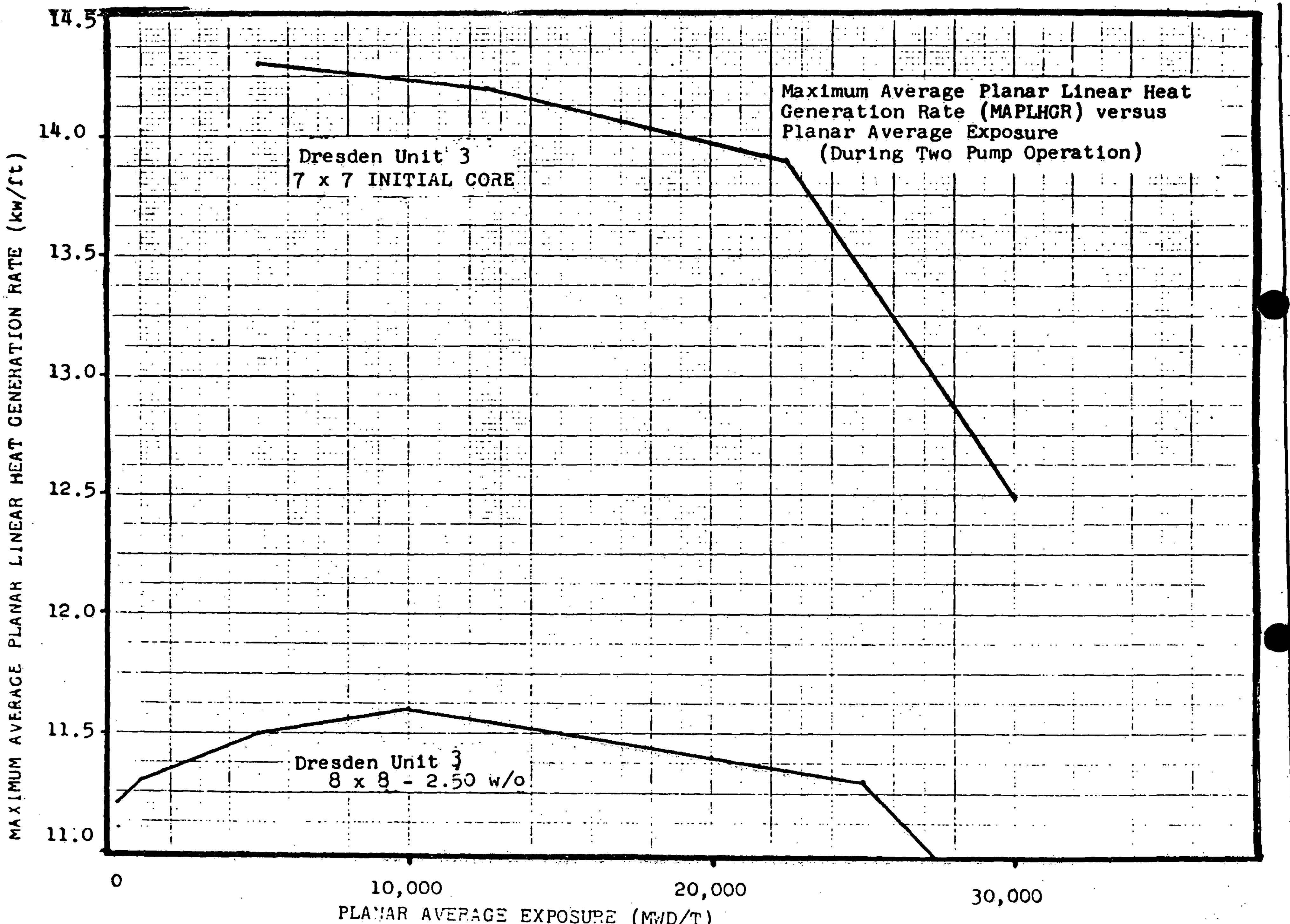
(1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)", NEDO-24046 August 1977.



MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (kw/ft)
14.5
14.0
13.5
13.0
12.5
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11.5
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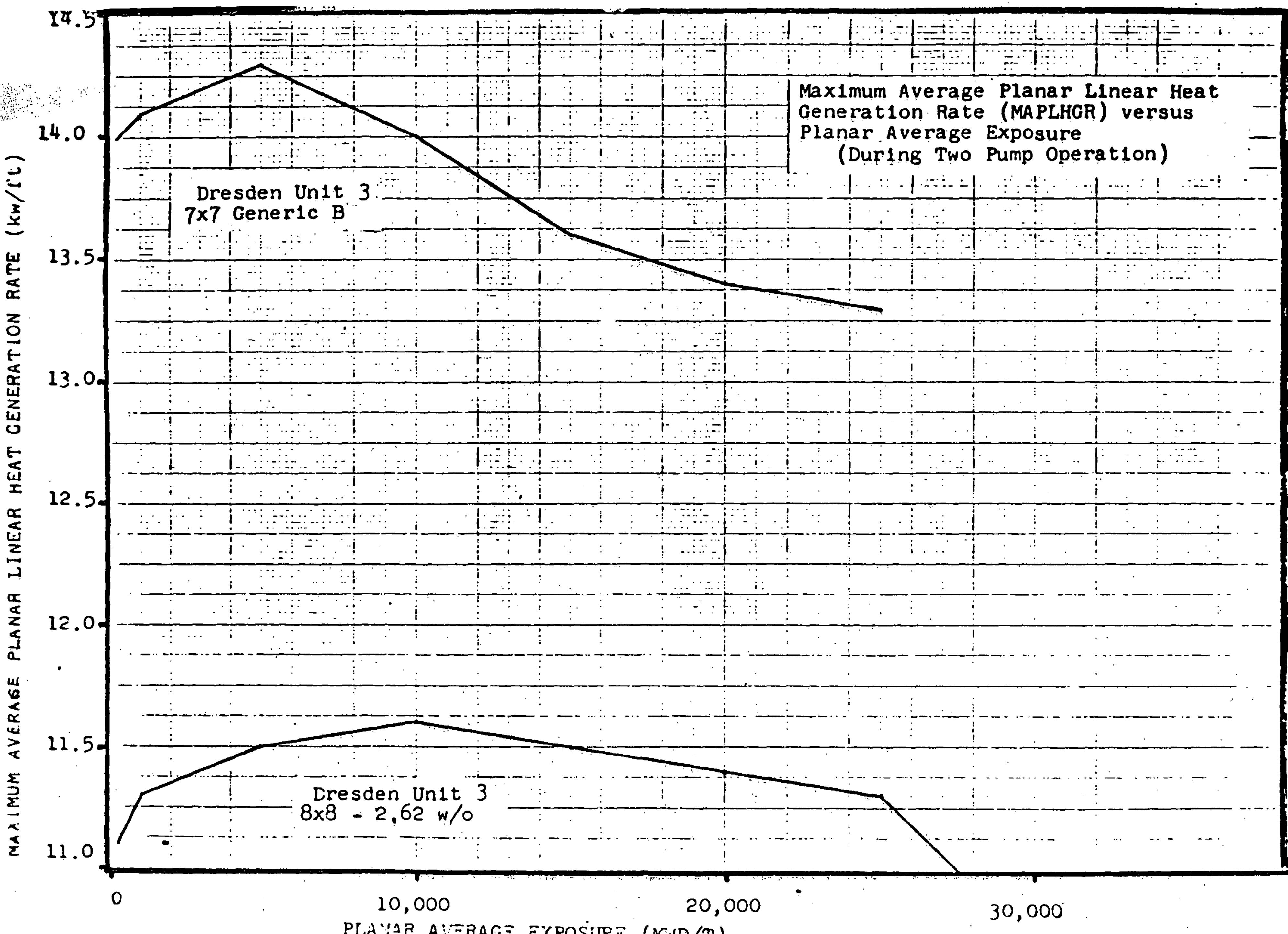
Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure (During Two Pump Operation)
Dresden Unit 3
7 x 7 INITIAL CORE
Dresden Unit 3
8 x 8 - 2.50 w/o
PLANAR AVERAGE EXPOSURE (MWD/T)

FIGURE 3.5.1.B



Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) versus Planar Average Exposure (During Two Pump Operation)
Dresden Unit 3
7x7 Generic B
Dresden Unit 3
8x8 - 2.62 w/o
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12.0
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10,000
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FIGURE 3.5.1.B

3.5 Limiting Conditions for Operation Bases

A. Core Spray and LPCI Mode of the RHR System - This specification assures that adequate emergency cooling capability is available.

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The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference (3). Using the results developed in this reference, the repair period is found to be less than 1/2 the test interval. This assumes that the core spray and LPCI subsystems constitute a 1 out of 3 system, however, the combined effect of the two systems to limit excessive clad temperatures must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 45 days and this specification is within this period. For multiple failures, a shorter interval is specified and to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in reference 3 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the

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(1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2,3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)", NEDO-24046, August 1977.

(2) NEDO-20566, General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50 Appendix K.

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3.5 Limiting Condition for Operation Bases (Cont'd)

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The peak cladding temperature following a postulated loss-of-coolant accident is primarily a function of the average LHGR of all the rods in a fuel assembly at any axial location and is only dependent secondarily on the rod to rod power distribution within a fuel assembly. Since expected local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ±20°F relative to the peak temperature for a typical fuel design, the limit on the average planar LHGR is sufficient to assure that calculated temperatures are below the 10CFR50, Appendix K limit.

The maximum average planar LHGRs shown in Figure 3.5.1 are based on calculations employing the models described in Reference (1). Power operation with LHGRs at or below those shown in Fig. 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit. Those values represent limits for operation to ensure conformance with 10CFR50 and Appendix K only if they are more limiting than other design parameters. The maximum average planar LHGRs plotted in Fig. 3.5.1 at higher exposures result in a calculated peak clad temperature of less than 2200°F. However the maximum average planar LHGRs are shown on Fig. 3.5.1 as limits because conformance calculations have not been performed to justify operation at LHGRs in excess of those shown.

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(1) "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)", NEDO-24046 August 1977.

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any rod is less than the design linear

ATTACHMENT II

QUAD-CITIES STATION UNITS 1 & 2

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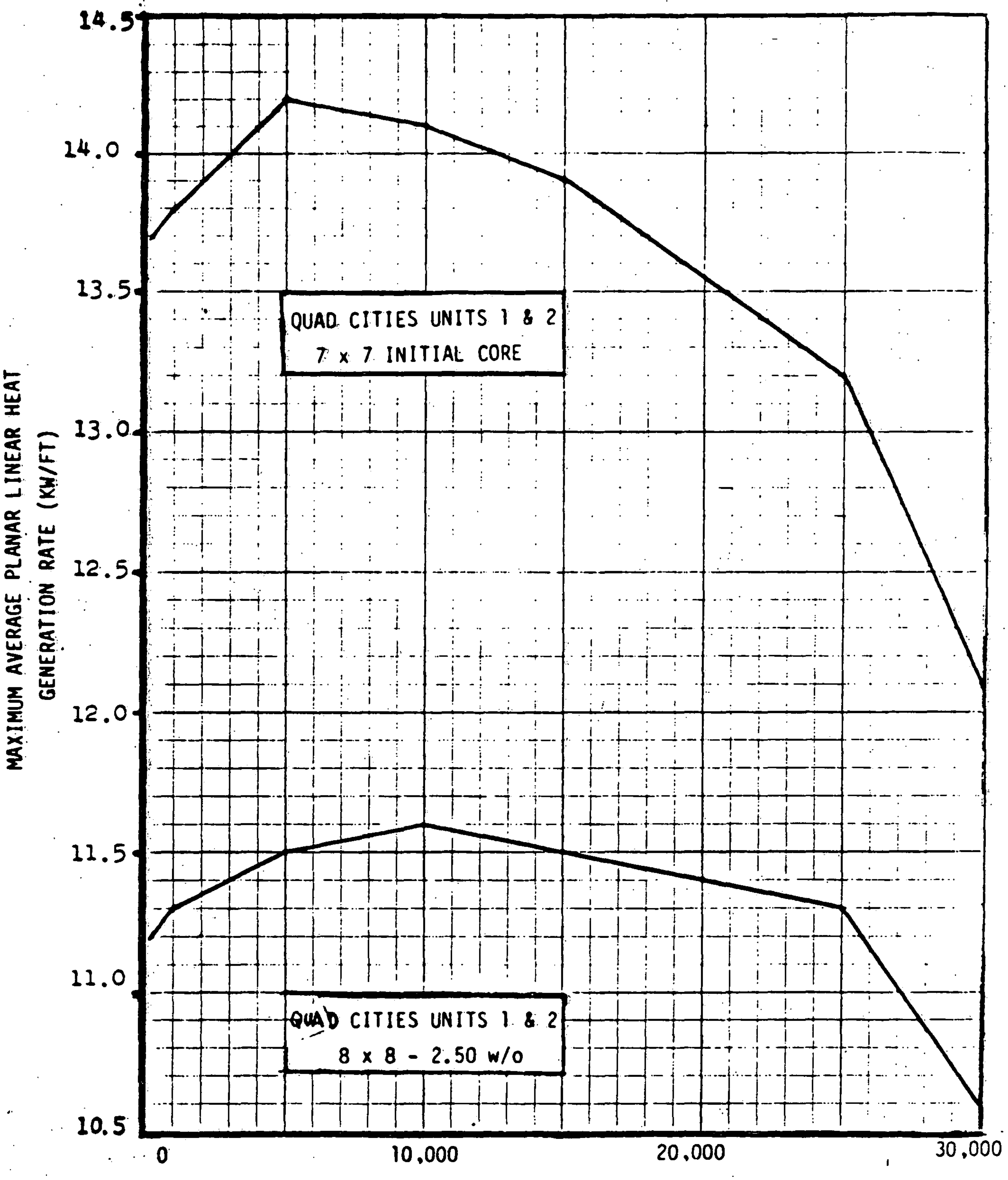

FIGURE 3.5-1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

(Sheet 1 of 3)

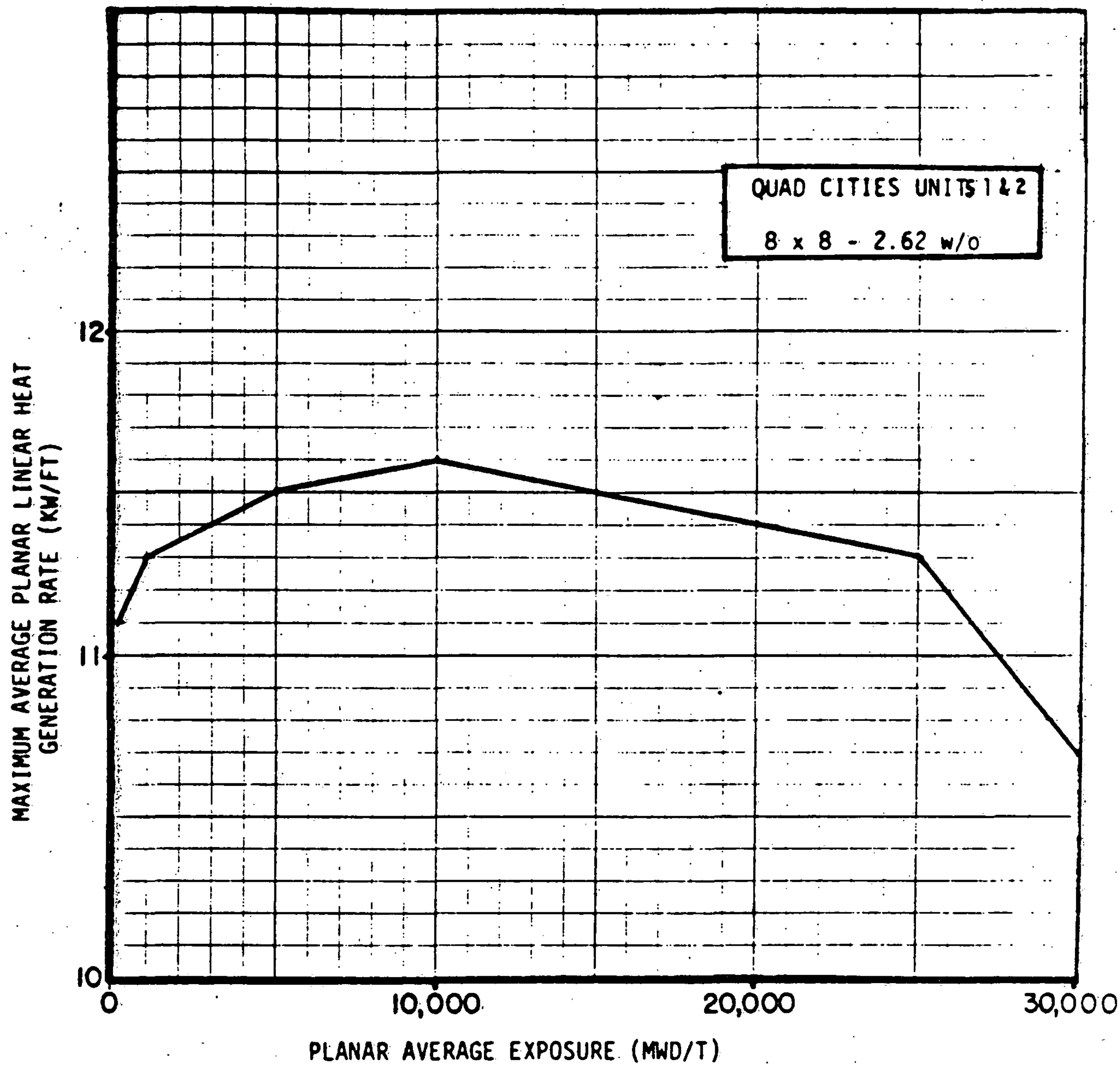

FIGURE 3.5-1 (Sheet 2 of 3)

MAXIMUM AVERAGE PLANAR
LINEAR HEAT GENERATION
RATE (MAPLHGR) VS.
PLANAR AVERAGE EXPOSURE

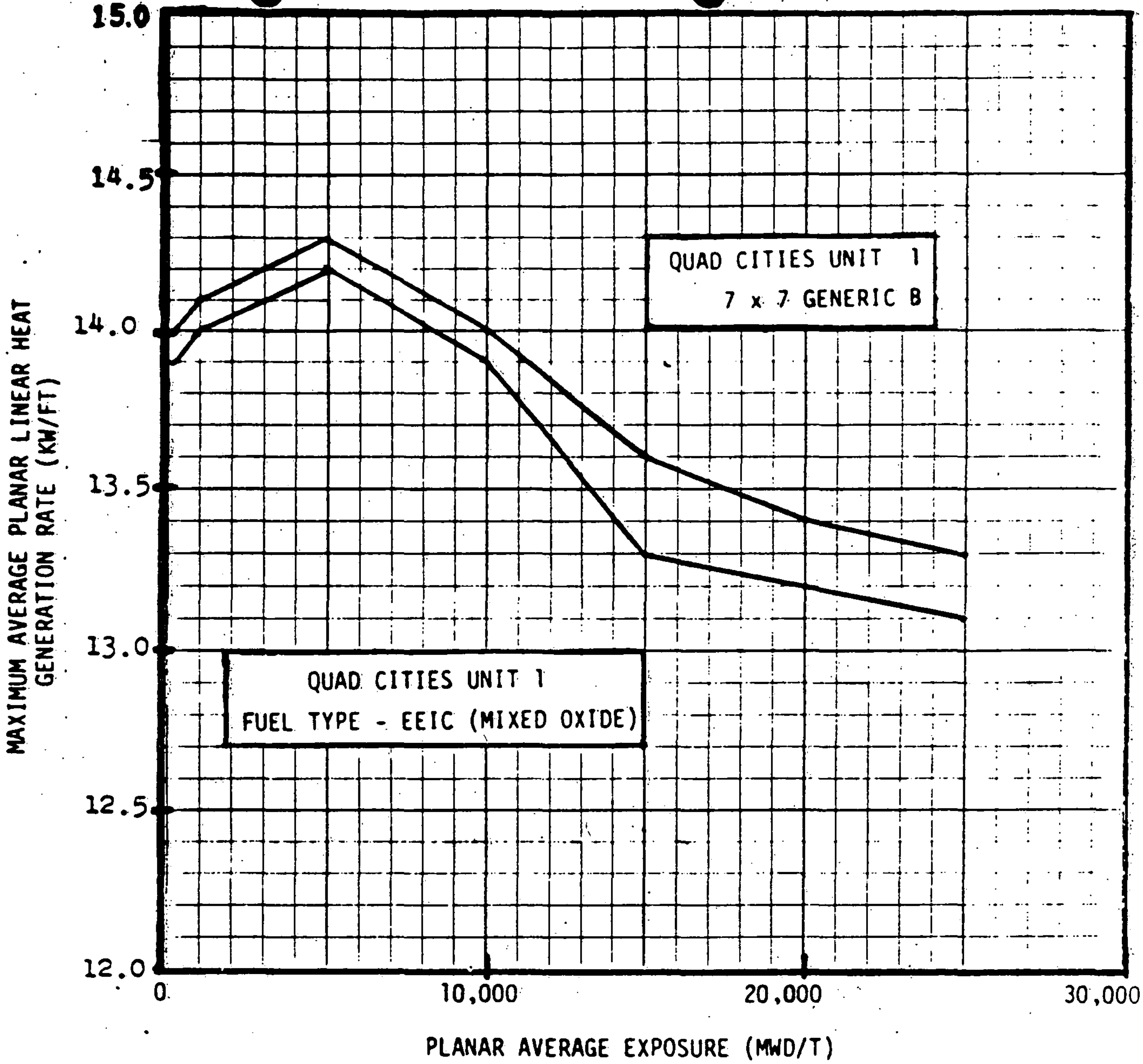

FIGURE 3.5-1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR VS. PLANAR AVERAGE EXPOSURE

(Sheet 3 of 3)

3.5 LIMITING CONDITION FOR OPERATION BASES

A. Core Spray and LPCI Mode of the RHR System

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

Based on the loss-of-coolant analytical methods described in General Electric Topical Report NEDO-20566 and the specific analysis in NEDO-24046, "Loss-of-Coolant Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)", August 1977, core cooling systems provide sufficient cooling to the core to dissipate the energy associated with the loss-of-coolant accident, to limit calculated fuel cladding temperature to less than 2200°F, to assure that core geometry remains intact, to limit cladding metal-water reaction to less than 1%, and to limit the calculated local metal-water reaction to less than 17%.

The limiting conditions of operation in Specifications 3.5.A.1 through 3.5.A.6 specify the combinations of operable subsystems to assure the availability of the minimum cooling systems noted above. No single failure of ECCS equipment occurring during a loss-of-coolant accident under these limiting conditions of operation will result in inadequate cooling of the reactor core.

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

The LPCI mode of the RHR system is designed to provide emergency cooling to the core by flooding in the event of a loss-of-coolant accident. This system functions in combination with the core spray system to prevent excessive fuel cladding temperature. The LPCI mode of the RHR system in combination with the core spray subsystem provides adequate cooling for break areas of approximately 0.2 ft² up to and including 4.18 ft², the latter being the double-ended recirculation line break with the equalizer line between the recirculation loops open without assistance from the high-pressure emergency core cooling subsystems.

The allowable repair times are established so that the average risk rate for repair would be no greater than the basic risk rate. The method and concept are described in Reference 1. Using the results developed in this reference, the repair period is found to be less than half the test interval. This assumes that the core spray subsystems and LPCI constitute a one-out-of-two system; however, the combined effect of the two systems to limit excessive cladding temperature must also be considered. The test interval specified in Specification 4.5 was 3 months. Therefore, an allowable repair period which maintains the basic risk considering single failures should be less than 30 days, and this specification is within this period. For multiple failures, a shorter interval is specified; to improve the assurance that the remaining systems will function, a daily test is called for. Although it is recognized that the information given in Reference 1 provides a quantitative method to estimate allowable repair times, the lack of operating data to support the analytical approach prevents complete acceptance of this method at this time. Therefore, the times stated in the specific items were established with due regard to judgment.

Should one core spray subsystem become inoperable, the remaining core spray subsystem and the entire LPCI mode of the RHR system are available should the need for core cooling arise. To assure that the remaining core spray, the LPCI mode of the RHR system, and the diesel generators are available, they are demonstrated to be operable immediately. This demonstration includes a manual initiation of the pumps and associated valves and diesel generators. Based on judgments of the reliability of the remaining systems, i.e., the core spray and LPCI, a 7-day repair period was obtained.

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

B. RHR Service Water

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

The containment cooling mode of the RHR system consists of two loops, each containing two RHR service water pumps, one heat exchanger, two RHR pumps, and the associated valves, piping, electrical equipment, and instrumentation. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability, as any one of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30-day repair period is adequate. Loss of one loop of the containment cooling mode of the RHR system leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the fact that when one loop of the containment cooling mode of the RHR system becomes inoperable, only one system remains, which is tested daily, a 7-day repair period was specified.

C. High-Pressure Coolant Injection

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

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

D. Automatic Pressure Relief

The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem or 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 or LPCI mode of the RHR system. The core spray subsystem and the LPCI mode of the RHR system provide sufficient flow of coolant to limit fuel cladding temperatures 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%.

Loss of 1 of the relief valves affects the pressure relieving capability and, therefore, a 7 day repair period is specified. Loss of more than one relief valve significantly reduces the pressure relief capability, thus a 24-hour repair period is specified based on the HPCI system availability during this period.

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.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This assures 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. I. M. Jacobs and P. W. Marritt, GE Topical Report APED-5736, 'Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards,' April 1969.
2. "Loss-of-Coolant-Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plant)", NEDO-24046, August 1977.
3. GE Topical Report NEDM-10735, 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' Section 3.2.1, Supplement 6, August 1973.
4. J. A. Hinds, GE, Letter to V. A. Moore, USAEC, 'Plant Evaluation with GE GEGAP-III,' December 12, 1973.
5. USAEC Report, 'Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels,' December 14, 1973.

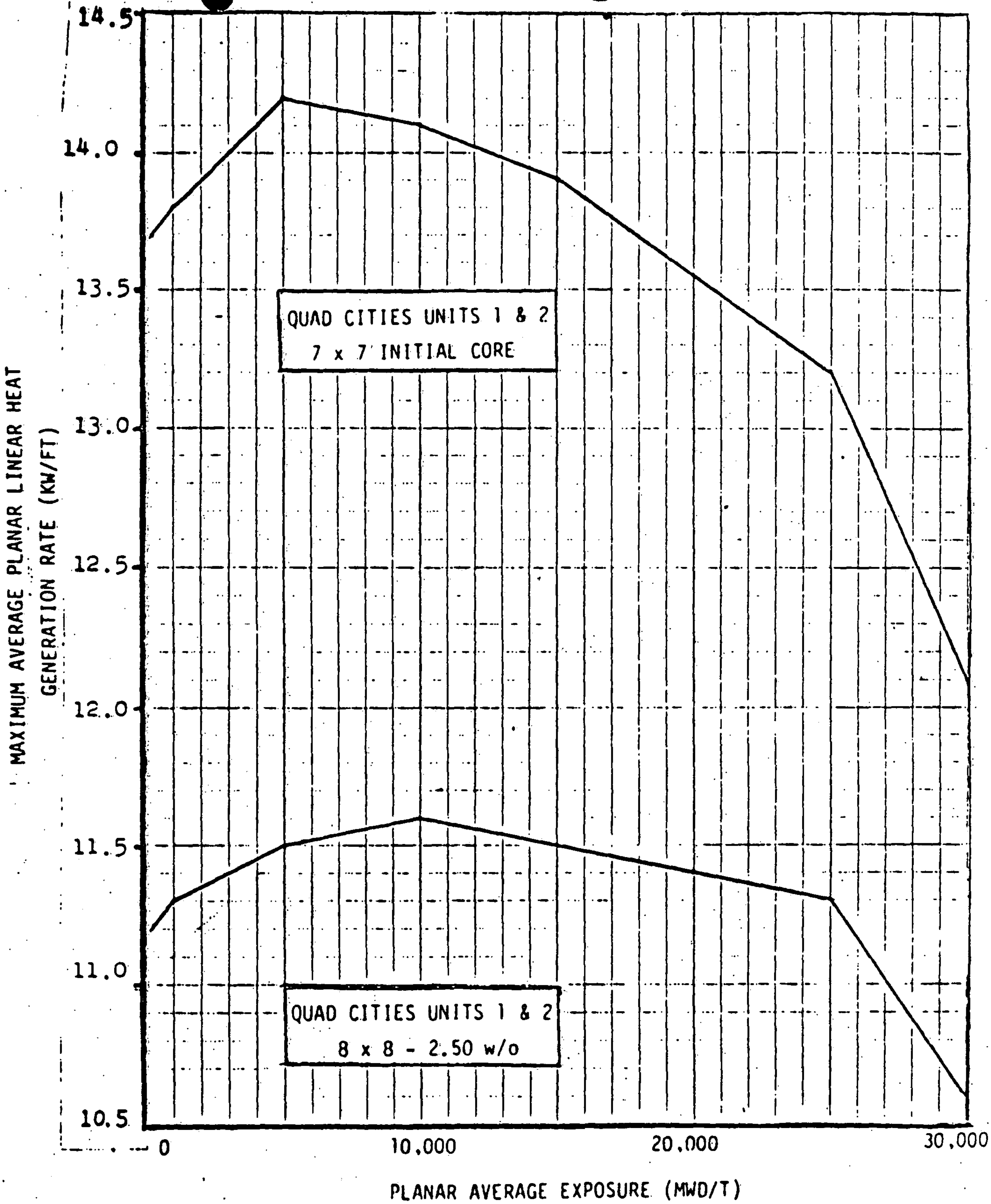

FIGURE 3.5-1

MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) VS. PLANAR AVERAGE EXPOSURE

(Sheet 1 of 2)

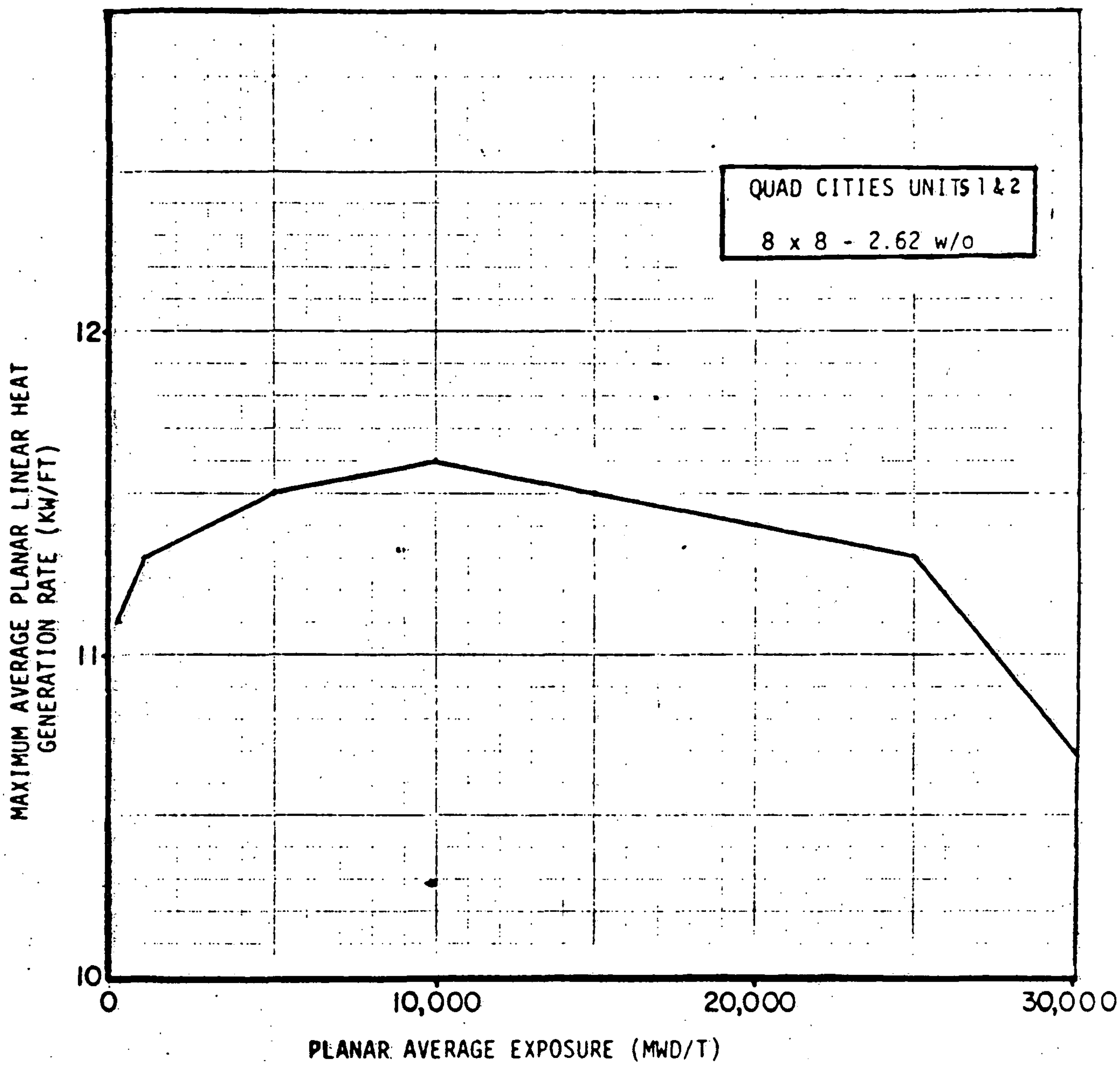

FIGURE 3.5-1 (Sheet 2 of 2)

MAXIMUM AVERAGE PLANAR
LINEAR HEAT GENERATION
RATE (MAPLHGR) VS.
PLANAR AVERAGE EXPOSURE

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.

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The relief valves of the automatic pressure relief subsystem are a backup to the HPCI subsystem. They enable the core spray subsystem or 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 or LPCI mode of the RHR system. The core spray subsystem and 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 remaining 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%.

Loss of 1 of the relief valves affects the pressure relieving capability and, therefore, a 7 day repair period is specified. Loss of more than one relief valve significantly reduces the pressure relief capability, thus a 24-hour repair period is specified based on the HPCI system availability during this period.

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 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 system. 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

H. Condensate Pump Room Flood Protection

See Specification 3.5.H.

I. Average Planar LHGR

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

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

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 specified is based on that presented in Reference 3 and assumes a linearly increasing variation in axial gaps between core bottom and top and assures with a 95% confidence that no more than one fuel rod exceeds the design linear heat-generation rate due to power spiking. An irradiation growth factor of 0.25% was used as the basis for determining Δ/P in accordance with References 4 and 5.

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, an MCPR of 1.18, 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 limiting value of MCPR stated in this specification is conservatively assumed to exist prior to the initiation of the transients. 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) 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 cycles reload licensing submittal 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 as the most restrictive over the entire cycle for each fuel type.

For core flow rates less than rated, the steady state MCPR is increased by the formula given in the specification. This assures that the MCPR will be maintained greater than that specified in Specification 1.1.A even in the event that the motor-generator set speed controller causes the scoop tube positioner for the fluid coupler to move to the maximum speed position.

References

1. "Loss-of-Coolant Accident Analysis Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations (Lead Plants)", NEDO-24046, August 1977.
2. GE Topical Report NEDO-20566, 'General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K.'
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. 'Fuel Densification Effects on General Electric Boiling Water Reactor Fuel,' Section 3.2.1, Supplement 6, August 1973.
5. J. A. Hinds, GE, Letter to V. A. Moore, USAEC, 'Plant Evaluation with GE GEGAP-III,' December 1973.
6. USAEC Report, 'Supplement 1 to the Technical Report on Densification of General Electric Reactor Fuels,' December 14, 1973.