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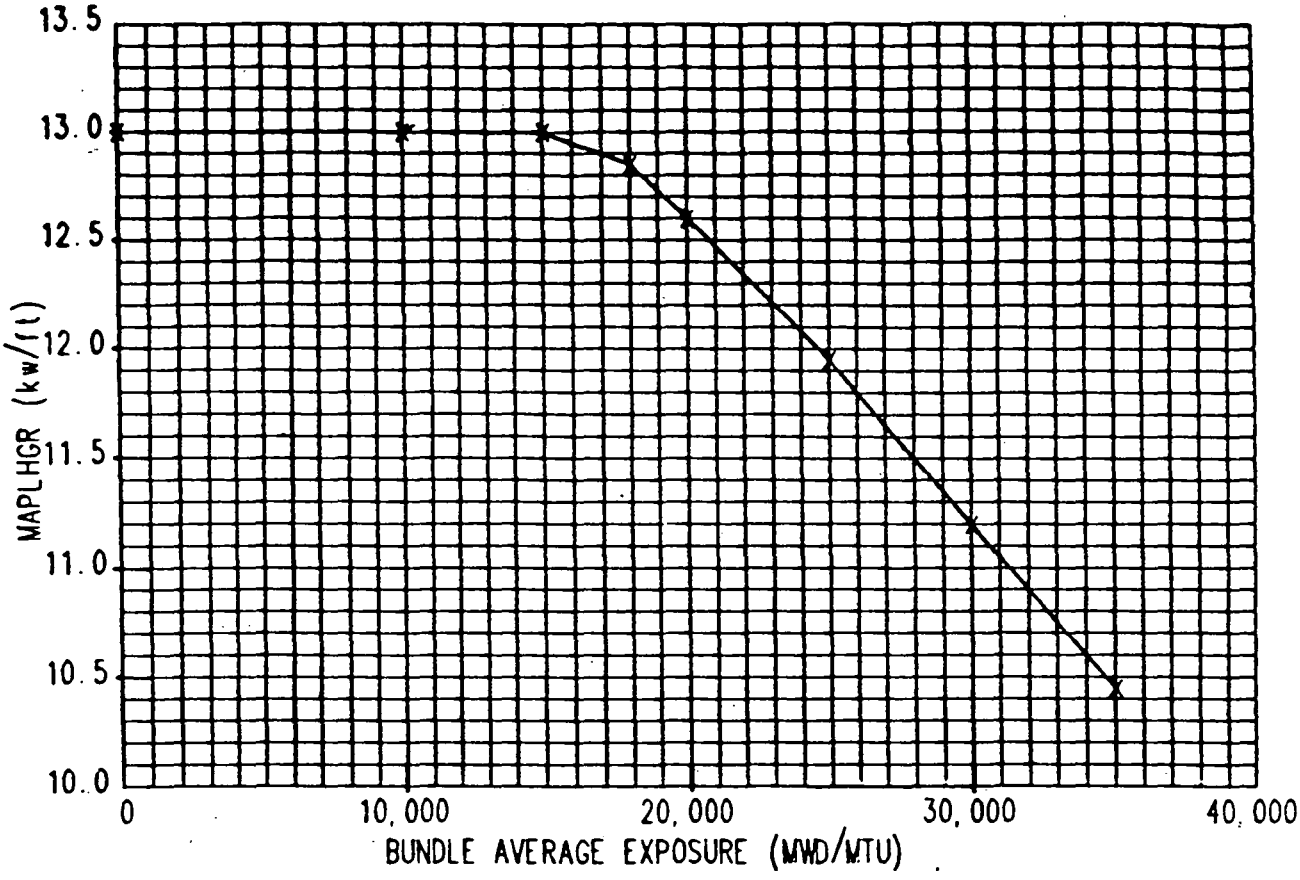
List of Figures

MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ENC 8x8 FUEL



The above graph is based on the following MAPLHGR summary for ENC 8x8 fuel design:

Bundle Average Exposure (MWD/MTU)	MAPLHGR Limit (kw/ft)
0	13.0
10,000	13.0
15,000	13.0
18,000	12.85
20,000	12.60
25,000	11.95
30,000	11.20
35,000	10.45

Figure 3.5-1
(Sheet 1 of 6)

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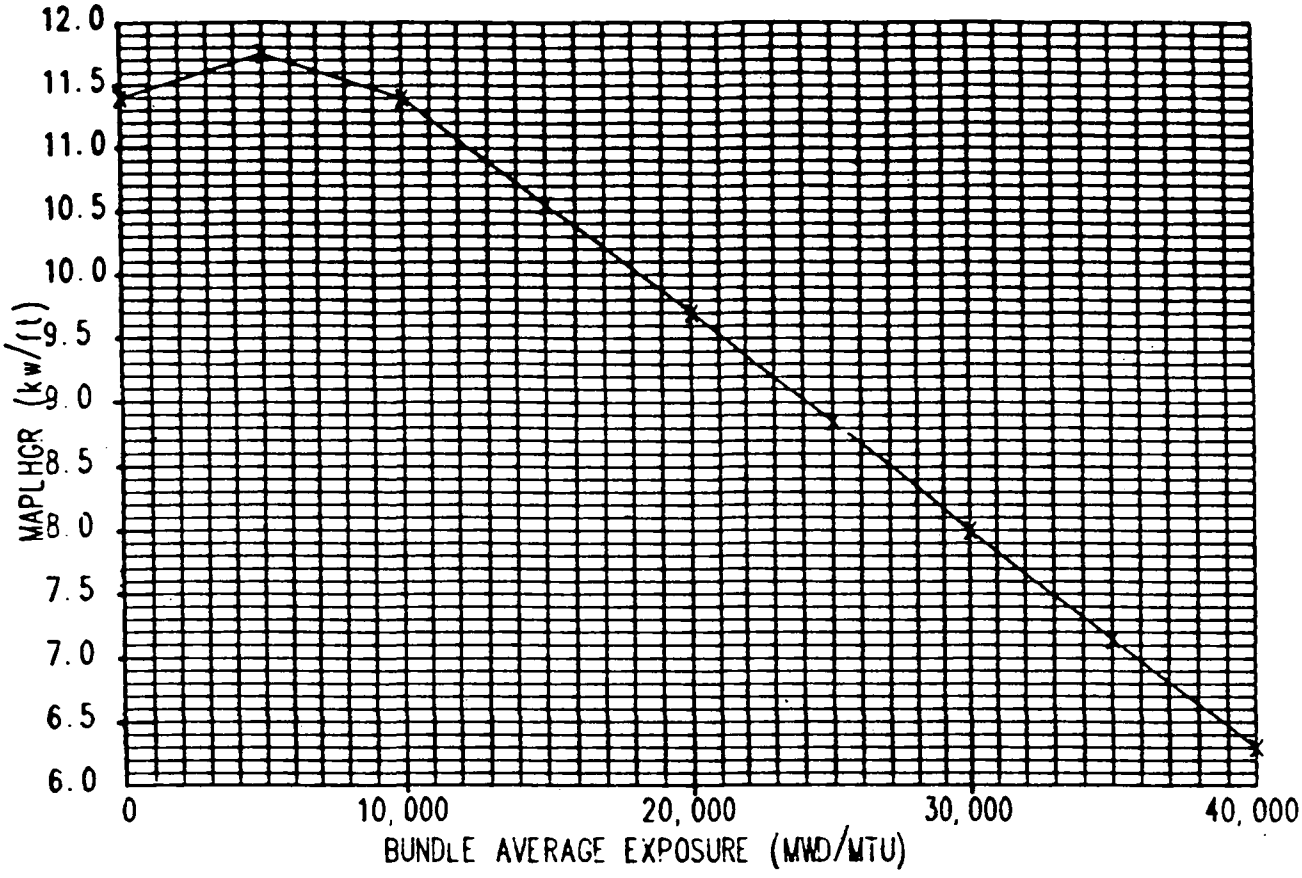
MAPLHGR LIMIT VS. BUNDLE AVERAGE EXPOSURE
ENC 9x9 FUEL



The above graph is based on the following MAPLHGR summary for ENC 9x9 fuel design:

Bundle Average Exposure (MWD/MTU)	MAPLHGR Limit (kw/ft)
0	11.40
5,000	11.75
10,000	11.40
15,000	10.55
20,000	9.70
25,000	8.85
30,000	8.00
35,000	7.15
40,000	6.30

Figure 3.5-1
(Sheet 2 of 6)

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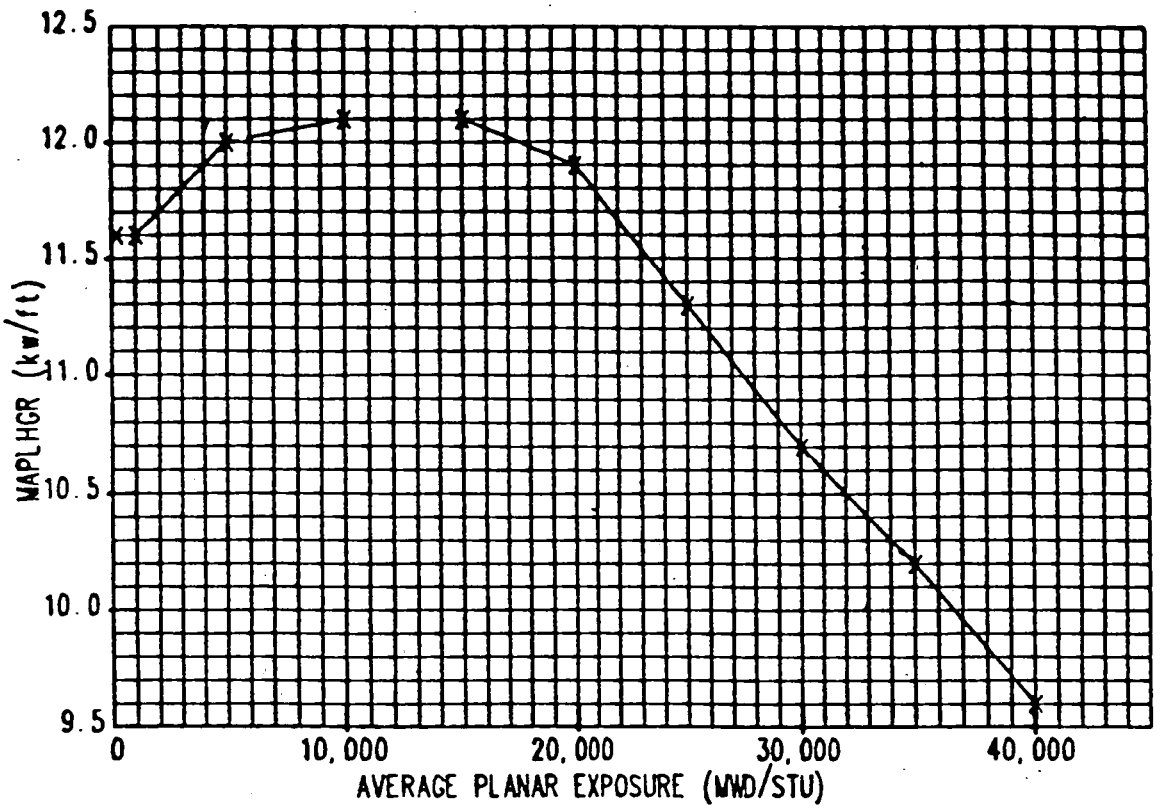
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB265L



The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB265L.

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.6
1,000	11.6
5,000	12.0
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6

Figure 3.5-1
(Sheet 3 of 6)

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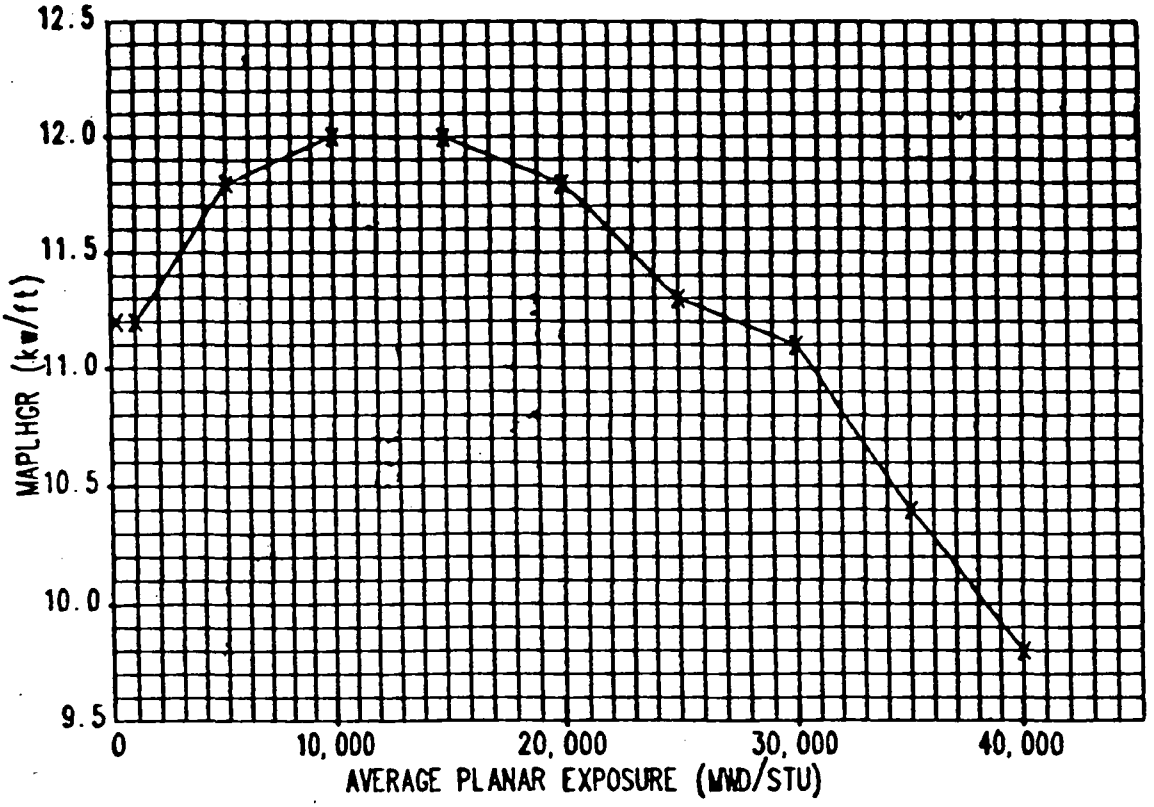
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB282



The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB282.

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.2
1,000	11.2
5,000	11.8
10,000	12.0
15,000	12.0
20,000	11.8
25,000	11.3
30,000	11.1
35,000	10.4
40,000	9.8

Figure 3.5-1
(Sheet 4 of 6)

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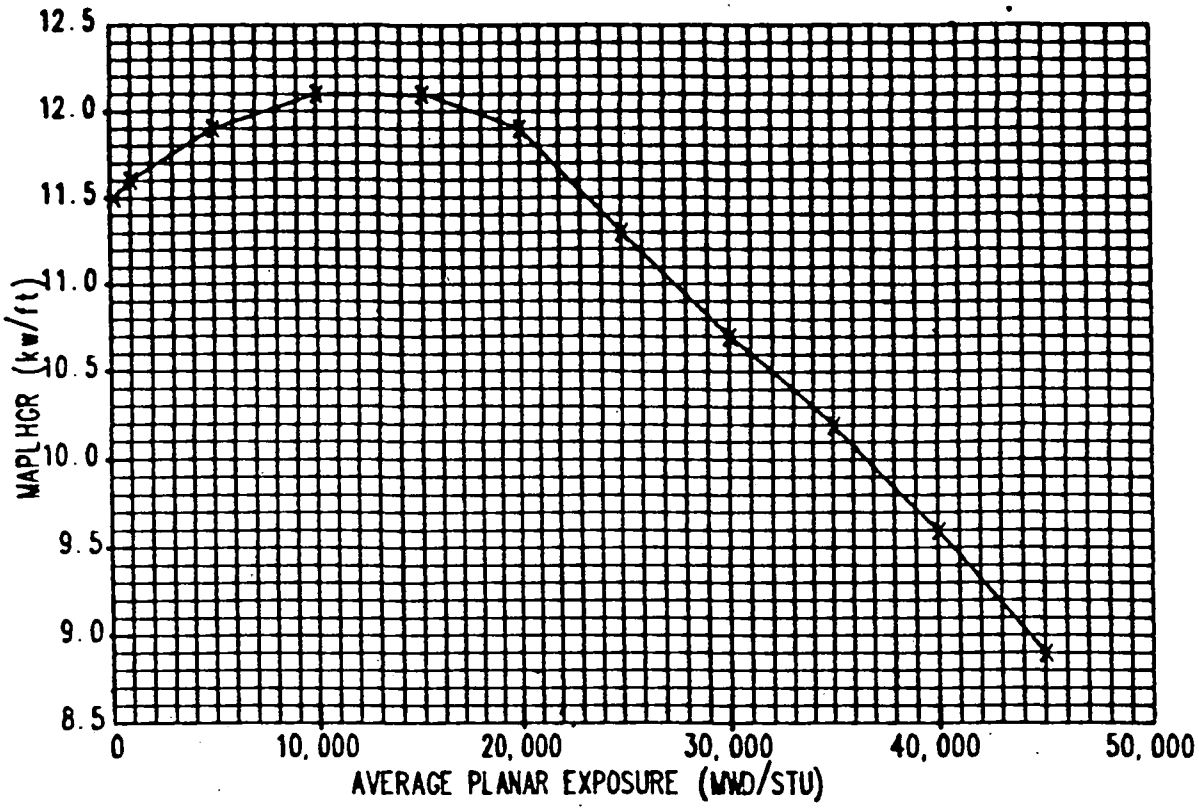
MAPLHGR LIMIT VS. AVERAGE PLANAR EXPOSURE
GE FUEL TYPE P8DRB265H



The above graph is based on the following MAPLHGR summary for GE fuel type P8DRB265H

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	12.1
20,000	11.9
25,000	11.3
30,000	10.7
35,000	10.2
40,000	9.6
45,000	8.9

Figure 3.5-1
(Sheet 5 of 6)

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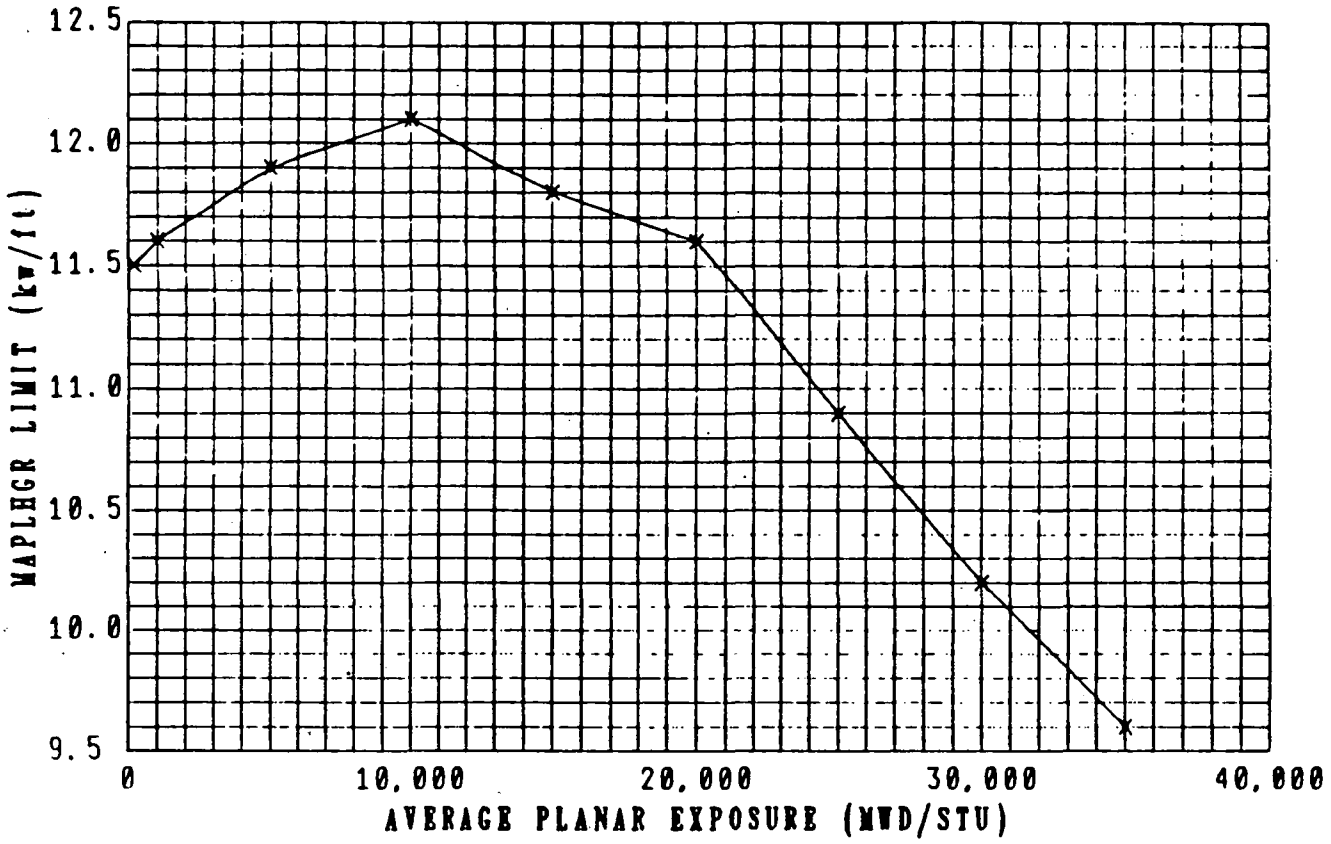
MAPLHGR VS. Average Planar Exposure
GE 8X8 LTA - Bundle LY5458



The above graph is based on the following MAPLHGR summary for the GE LTA, bundle LY5458:

Average Planar Exposure (MWD/STU)	MAPLHGR Limit (kw/ft)
200	11.5
1,000	11.6
5,000	11.9
10,000	12.1
15,000	11.8
20,000	11.6
25,000	10.9
30,000	10.2
35,000	9.6

Figure 3.5-1
(Sheet 6 of 6)

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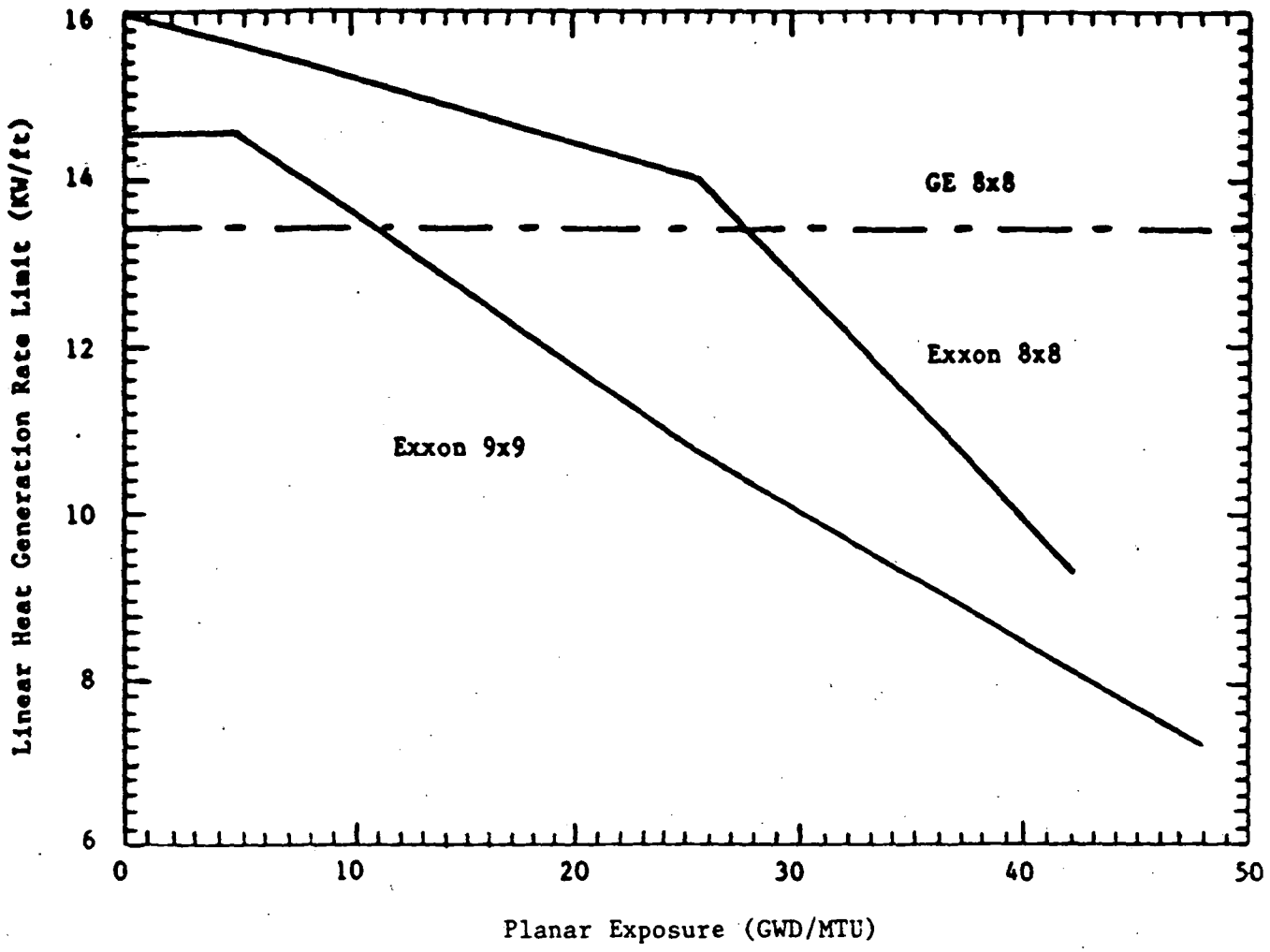

Exxon 8x8 Fuel	
Exposure	LHGR
0.00	16.00
25.40	14.10
42.00	9.30

Exxon 9x9 Fuel	
Exposure	LHGR
0.00	14.50
5.00	14.50
25.20	10.80
48.00	7.20

Figure 3.5-1A

LINEAR HEAT GENERATION RATE VS.
NODAL EXPOSURE

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

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

1.32 for Exxon 9x9 fuel

1.31 for GE and Exxon 8x8 fuel

For core flows other than rated, the MCPR Operating Limit shall be as follows:

1. Manual Flow Control - the MCPR Operating Limit shall be the value from Figure 3.5-2 sheet 1 or the above rated flow value, whichever is greater.

2. Automatic Flow Control - the MCPR Operating Limit is the greatest of the following:
   a. The above rated flow value;
   b. The value from Figure 3.5-2 sheet 1; or
   c. The interpolated value from Figure 3.5-2 sheets 2 and 3.

3. During Single Loop Operation, all the rated flow MCPR operating limits shall be increased by an additive factor of 0.03.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

K. Minimum Critical Power Ratio (MCPR)

MCPR shall be determined daily during a reactor power operation at greater than or equal to 25% rated thermal power and following any change in power level or distribution that would cause operation with a limiting control rod pattern as described in the bases for Specification 3.3.B.5.

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

If at any time during steady state power operation, it is determined 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 two (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.

In the event the average 90% scram insertion time determined by Specification 3.3.C for all operable control rods exceeds 2.77 seconds, the MCPR operating limit shall be increased by adding the amount equal to [0.238T - 0.66] where T equals the average 90% scram insertion time for the most recent half-core or full core surveillance data from Specification 4.3.C. Consequentially, the Automatic Flow Control MCPR Operating Limit must also be evaluated in accordance with Specification 3.5.K.2.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

L. Condensate Pump Room Flood Protection

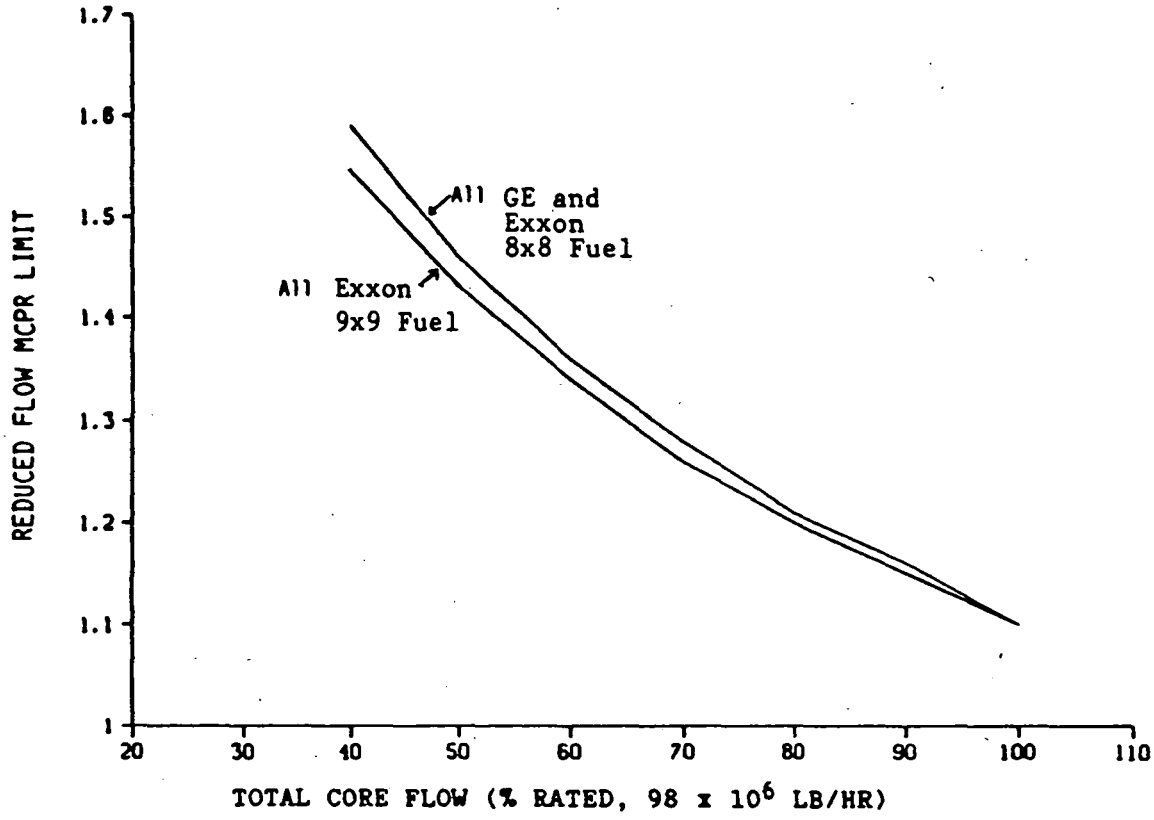
1. The system is installed to prevent or mitigate the consequences of flooding of the condensate pump room and shall be operable prior to startup of the reactor.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

L. Condensate Pump Room Flood Protection

1. The following surveillance requirements shall be observed to assure that the condensate pump room flood protection is operable.

a. The testable penetrations through the walls of CCSW pump vaults shall be checked during each operating cycle by pressurizing to 15 plus or minus 2 psig and checking for leaks using a soap bubble solution. The criteria for acceptance should be no visible leakage through the soap bubble solution. The bulkhead door shall be checked during each operating cycle by hydrostatically testing the door at 15 plus or minus 2 psig and checking to verify that leakage around the door is less than one gallon per hour.



The above curves are based on the following MCPR Limit summary for reduced Total Core Flow:

Total Core Flow (% Rated)	MCPR Limit GE and Exxon 8x8	MCPR Limit Exxon 9x9
100	1.10	1.10
90	1.16	1.15
80	1.21	1.20
70	1.28	1.26
60	1.36	1.34
50	1.46	1.43
40	1.59	1.55

Figure 3.5-2 (Sheet 1 of 3)
MCPR Limit for reduced Total Core Flow

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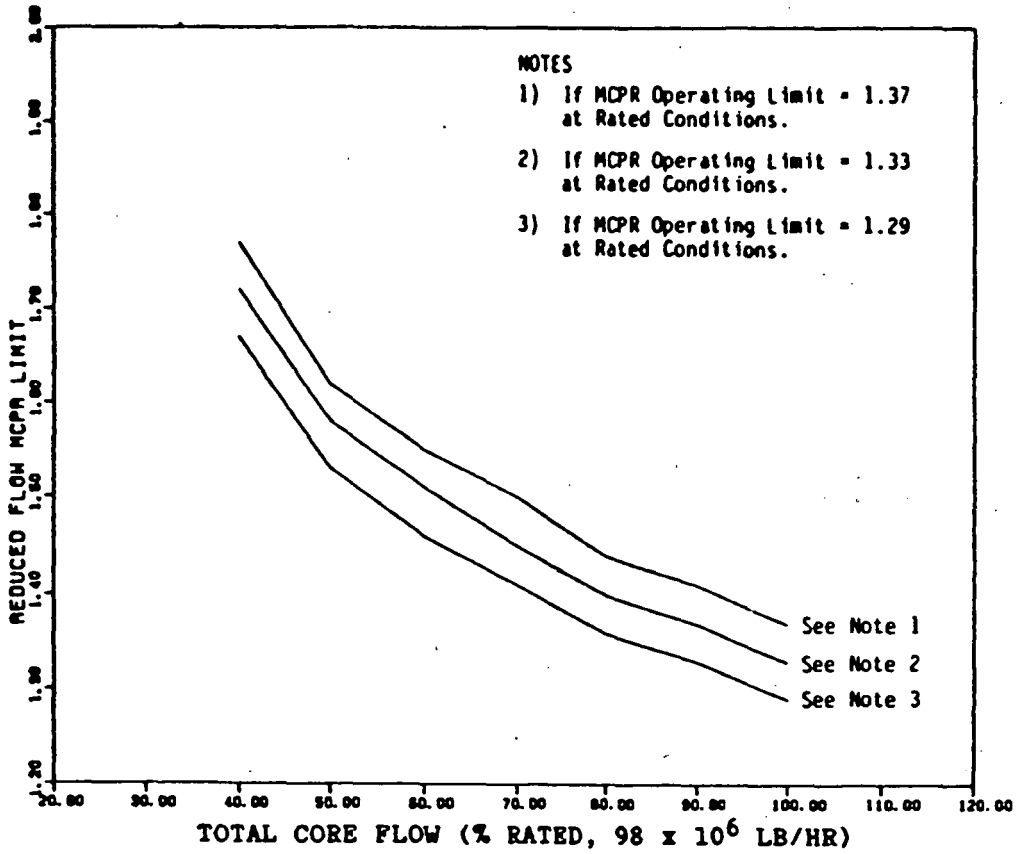

The above GE and Exxon 8x8 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

Total Core Flow (% Rated)	MCPR Operating Limit for GE and Exxon 8x8 fuel* 1.29	1.33	1.37
100	1.29	1.33	1.37
90	1.33	1.37	1.41
80	1.36	1.40	1.44
70	1.41	1.45	1.50
60	1.46	1.51	1.55
50	1.53	1.58	1.62
40	1.67	1.72	1.77

* Column headers are MCPR Operating Limits at rated flow.

Figure 3.5-2 (Sheet 2 of 3)
GE and Exxon 8x8 MCPR Operating Limit For Automatic Flow Control

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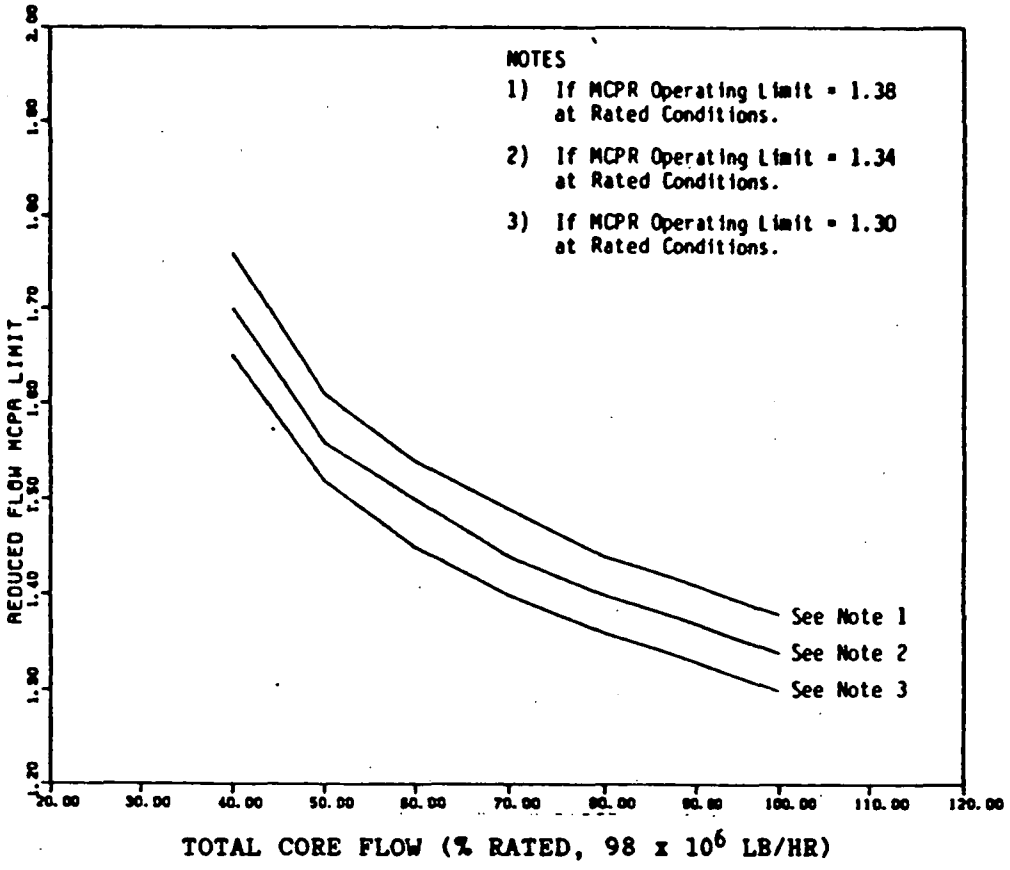

The above Exxon 9x9 curves are based on the following MCPR operating limit summary for Automatic Flow Control:

Total Core Flow (% Rated)	MCPR Operating Limit for Exxon 9x9 fuel* 1.30	1.34	1.38
100	1.30	1.34	1.38
90	1.33	1.37	1.41
80	1.36	1.40	1.44
70	1.40	1.44	1.49
60	1.45	1.50	1.54
50	1.52	1.56	1.61
40	1.65	1.70	1.76

* Column headers are MCPR Operating Limits at rated flow.

Figure 3.5-2 (Sheet 3 of 3)
Exxon 9x9 MCPR Operating Limit For Automatic Flow Control

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

b. The CCSW Vault Floor drain shall be checked during each operating cycle by assuring that water can be run through the drain line and actuating the air operated valves by operation of the following sensor:

   i. loss of air

   ii. high level in the condensate pump room (5'0")

c. The condenser pit five foot trip shall have a trip setting of less than or equal to five feet zero inches. The five foot trip circuit for each channel shall be checked once every three months. The 3 and 1 foot alarms shall have a setting of less than or equal to three feet zero inches and less than or equal to 1 foot 0 inches. A logic system functional test, including all alarms, shall be performed during the refueling outage.

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3.5 LIMITING CONDITION FOR OPERATION (Cont'd.)

2. The condenser pit water level switches shall trip the condenser circulating water pumps and alarm in the control room if water level in the condenser pit exceeds a level of 5 feet above the pit floor. If a failure occurs in one of these trip and alarm circuits, the failed circuit shall be immediately placed in a trip condition and reactor operation shall be permissible for the following seven days unless the circuit is sooner made operable.

3. If Specification 3.5.L.1 and 2 cannot be met, reactor startup shall not commence or if operating, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 24 hours.

4.5 SURVEILLANCE REQUIREMENT (Cont'd.)

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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.

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

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(1) "Loss of Coolant Accident Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revisions 1, April 1979.

(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.)

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 judgement.

Should one core spray subsystem become inoperable, the remaining core spray and the entire LPCI system are available should the reactor core cooling arise. To assure that the remaining core spray and LPCI subsystems 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 judgements 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 LPCI pump occur, a nearly full complement of core and containment cooling equipment is available. Three LPCI 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 subsystem is not available, at least 2 LPCI pumps must be available to fulfill the containment cooling function. The 7-day repair period is set on this basis.

B. Containment Cooling Service Water - The containment heat removal portion of the LPCI/containment cooling subsystem 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, therefore, is more than ample to provide the required heat removal capability. (Ref. Section 5.2.3.2 SAR).

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger and 2 LPCI pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one containment cooling service water pump does not seriously jeopardize the containment cooling capability as any 2 of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left a 30-day repair period is adequate. Loss

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

of 1 containment cooling subsystem 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 facts that when one containment cooling subsystem 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 or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site electrical power. For the pipe breaks for which the HPCI is intended to function the core never uncovers and is continuously cooled and thus no clad damage occurs. (Ref. Section 6.2.5.3 SAR). 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 back-up to the HPCI subsystem. They enable the core spray or LPCI 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 sprays or LPCI. The core spray and/or LPCI provide sufficient flow of coolant to adequately cool the core.

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 1 relief valve significantly reduces the pressure relief capability and thus a 24-hour repair period is specified.

E. Isolation Cooling System - The turbine main condenser is normally available. The isolation condenser is provided for core decay heat removal following reactor isolation and scram. The isolation condenser has a heat removal capacity sufficient to handle the decay heat production at 300 seconds following a scram. Water will be lost from the reactor vessel through the relief valves in the 300 seconds following isolation and scram. This represents a minor loss relative to the vessel inventory.

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3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

The system may be manually initiated at any time. The system is automatically initiated on high reactor pressure in excess of 1060 psig sustained for 15 seconds. The time delay is provided to prevent unnecessary actuation of the system during anticipated turbine trips. Automatic initiation is provided to minimize the coolant loss following isolation from the main condenser. To be considered operable the shell side of the isolation condenser must contain at least 11,300 gallons of water. Make-up water to the shell side of the isolation condenser is provided by the condensate transfer pumps from the condensate storage tank. The condensate transfer pumps are operable from on-site power. The fire protection system is also available as make-up water. An alternate method of cooling the core upon isolation from the main condenser is by using the relief valves and HPCI subsystem in a feed and bleed manner. Therefore, the high pressure relief function and the HPCI must be available together to cope with an anticipated transient so the LCO for HPCI and relief valves is set upon this function rather than their function as depressurization means for a small pipe break.

F. Emergency Cooling Availability - The purpose of Specification D 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 2 LPCI pumps would be available. Likewise, if 2 LPCI pumps were out of service and 2 containment 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.

Dresden Units 2 and 3 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 diesel generators. All of these systems have been sized to perform their intended function considering the simultaneous operation of both units.

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3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For the safety related shared features of each plant, the Technical Specifications for that unit contain the operability and surveillance requirements for the shared feature; thus, the level of operability for one unit is maintained independently of the status of the other. For example, the shared diesel (2/3 diesel) would be mentioned in the specifications for both Units 2 and 3 and even if Unit 3 were in the Cold Shutdown Condition and needed no diesel power, readiness of the 2/3 diesel would be required for continuing Unit 2 operation.

G. Specification 3.5.F.4 provides that should this occur, no work will be performed which could preclude adequate emergency cooling capability being available. Work is prohibited unless it is in accordance with specified procedures which limit the period that the control rod drive housing is open and assures that the worst possible loss of coolant resulting from the work will not result in uncovering the reactor core. Thus, this specification assures adequate core cooling. Specification 3.9 must be consulted to determine other requirements for the diesel generator.

Specification 3.5.F.5 provides assurance that an adequate supply of coolant water is immediately available to the low pressure core cooling systems and that the core will remain covered in the event of a loss of coolant accident while the reactor is depressurized with the head removed.

H. Maintenance of Filled Discharge Pipe - If the discharge piping of the core spray, LPCI, and HPCI are not filled, a water hammer can develop in this piping when the pump and/or pumps are started.

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 affects 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

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3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

local variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than plus or minus 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 References (1), (2) and (3). Power operation with APLHGRs at or below those shown in Figure 3.5.1 assures that the peak cladding temperature following a postulated loss-of-coolant accident will not exceed the 2200°F limit.

General Electric has analyzed the effects that Single Loop Operation has on LOCA events (Reference 4). For breaks in the idle loop, the above Dual Loop Operation results are conservative (Reference 1). For breaks in the active loop, the event is more severe primarily due to a more rapid loss of core flow. By decreasing the results of the previous analyses to 70% of the original value, all applicable criteria are met. ENC concurs with GE that the reduction factor is conservatively applicable for cores fueled with 8x8 and 9x9 fuel (Reference 5).

The maximum average planar LHGRs for G.E. fuel plotted in Figure 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 Figure 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 Analyses Report for Dresden Units 2, 3 and Quad-Cities Units 1, 2 Nuclear Power Stations," NEDO-24146A, Revision 1, April, 1979.

(2) XN-NF-82-88 "Dresden Unit 2 LOCA Analysis Using the ENC EXEM/BWR Evaluation Model MAPLHGR Results"

(3) XN-NF-85-63 "Dresden Unit 3 LOCA-ECCS Analysis MAPLHGR results for 9x9 fuel", dated September 1985.

(4) NEDO-24807, "Dresden Nuclear Power Station Units 2 and 3 and Quad Cities Nuclear Power Station, Units 1 and 2 Single Loop Operation," dated December 1980.

(5) XN-NF-86-103 "Dresden Unit 2 cycle 11 Reload Analysis" dated September 1986.

3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

J. Local LHGR

This specification assures that the maximum linear heat generation rate in any fuel rod is less than the design linear heat generation rate even if fuel pellet densification is postulated.

K. Minimum Critical Power Ratio (MCPR)

The steady-state values for MCPR specified in the Specification were determined using the THERMEX thermal limits methodology described in XN-NF-80-19, Volume 3. The safety limit implicit in the Operating limits is established so that during sustained operation at the MCPR safety limit, at least 99.9% of the fuel rods in the core are expected to avoid boiling transition. The Limiting Transient delta CPR implicit in the operating limits was calculated such that the occurrence of the limiting transient from the operating limit will not result in violation of the MCPR safety limit in at least 95% of the random statistical combinations of uncertainties.

Transient events of each type anticipated during operation of a BWR/3 were evaluated to determine which is most restrictive in terms of thermal margin requirements. The generator load rejection/turbine trip without bypass is typically the limiting event. The thermal margin effects of the event are evaluated with the THERMEX Methodology and appropriate MCPR limits consistent with the XN-3 critical power correlation are determined. Several factors influence which transient results in the largest reduction in critical power ratio, such as the cycle-specific fuel loading, exposure and fuel type. The current cycle's reload licensing analyses identifies the limiting transient for that cycle.

As described in Specification 4.3.C.3 and the associated Bases, observed plant data were used to determine the average scram performance used in the transient analyses for determining the MCPR Operating Limit. If the current cycle scram time performance falls outside of the distribution assumed in the analyses, an adjustment of the MCPR limit may be required to maintain margin to the MCPR Safety Limit during transients. Compliance with the assumed distribution and adjustment of the MCPR Operating Limit will be performed in accordance with Technical Specifications 4.3.C.3. and 3.5.K.

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3.5 LIMITING CONDITION FOR OPERATION BASES (Cont'd.)

For core flows less than rated, the MCPR Operating Limit established in the specification is adjusted to provide protection of the MCPR Safety Limit in the event of an uncontrolled recirculation flow increase to the physical limit of pump flow. This protection is provided for manual and automatic flow control by choosing the MCPR operating limit as the value from Figure 3.5-2 Sheet 1 or the rated core flow value, whichever is greater. For Automatic Flow Control, in addition to protecting the MCPR Safety Limit during the flow run-up event, protection is provided against violating the rated flow MCPR Operating Limit during an automatic flow increase to rated core flow. This protection is provided by the reduced flow MCPR limits shown in Figure 3.5-2 Sheet 2 or Sheet 3 where the curve corresponding to the current rated flow MCPR limit is used (linear interpolation between the MCPR limit lines depicted is permissible). Therefore, for Automatic Flow Control, the MCPR Operating Limit is chosen as the value from Figure 3.5-2 Sheet 1, Sheet 2, Sheet 3 or the rated flow value, whichever is greatest. It should be noted that if the rated flow MCPR Limit must be increased due to degradation of control rod scram times during the current cycle, the new value of the rated flow MCPR limit is applied when using Figure 3.5-2 Sheets 2 and 3.

Analyses have demonstrated that transient events in Single Loop Operation are bounded by those at rated conditions; however, due to the increase in the MCPR fuel cladding integrity safety limit in Single Loop Operation, an equivalent adder must be uniformly applied to all MCPR LCO to maintain the same margins to the MCPR fuel cladding integrity safety limit.

L. Flood Protection

Condensate pump room flood protection will assure the availability of the containment cooling service water system (CCSW) during a postulated incident of flooding in the turbine building. The redundant level switches in the condenser pit will preclude any postulated flooding of the turbine building to an elevation above river water level. The level switches provide alarm and circulating water pump trips in the event a water level is detected in the condenser pit.

4.5 SURVEILLANCE REQUIREMENT BASES

(A thru F)

The testing interval for the core and containment cooling systems is based on quantitative reliability analysis, judgement and practicality. The core cooling systems have not been designed to be fully testable during operation. For example the core spray final admission valves do not open until reactor pressure has fallen to 350 psig thus during operation even if high drywell pressure were stimulated the final valves would not open. In the case of the HPCI, automatic initiation during power operation would result in pumping cold water into the reactor vessel which is not desirable.

The systems can be automatically actuated during a refueling outage and this will be done. To increase the availability of the individual components of the core and containment cooling systems the components which make up the system i.e., instrumentation, pumps, valve operators, etc., are tested more frequently. The instrumentation is functionally tested each month. Likewise the pumps and motor-operated valves are also tested each month to assure their operability. The combination of a yearly simulated automatic actuation test and monthly tests of the pumps and valve operators is deemed to be adequate testing of these systems.

With components or subsystems out-of-service overall core and containment cooling reliability is maintained by demonstrating the operability of the remaining cooling equipment. The degree of operability to be demonstrated depends on the nature of the reason for the out-of-service equipment. For routine out-of-service periods caused by preventative maintenance, etc., the pump and valve operability checks will be performed to demonstrate operability of the remaining components. However, if a failure, design deficiency, etc., caused the out-of-service period, then the demonstration of operability should be thorough enough to assure that a similar problem does not exist on the remaining components. For example, if an out-of-service period were caused by failure of a pump to deliver rated capacity due to a design deficiency, the other pumps of this type might be subjected to a flow rate test in addition to the operability checks.

The requirement of 180 psig at 3500 gpm at the containment cooling service water (CCSW) pump discharge provides adequate margin to ensure that the LPCI/CCSW system provides the design

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

bases cooling water flow and maintains 20 psig differential pressure at the containment cooling heat exchanger. This differential pressure precludes reactor coolant from entering the river water side of the containment cooling heat exchangers.

The verification of Main Steam Relief Valve operability during manual actuation surveillance testing must be made independent of temperatures indicated by thermocouples downstream of the relief valves. It has been found that a temperature increase may result with the valve still closed. This is due to steam being vented through the valve actuation mechanism during the surveillance test. By first opening a turbine bypass valve, and then observing its closure response during relief valve actuation, positive verification can be made for the relief valve opening and passing steam flow. Closure response of the turbine control valves during relief valve manual actuation would likewise serve as an adequate verification for relief valve opening. This test method may be performed over a wide range of reactor pressure greater than 150 psig. Valve operation below 150 psig is limited by the spring tension exhibited by the relief valves.

G. Deleted

H. Maintenance of Filled Discharge Pipe

The surveillance requirements to assure that the discharge piping of the core spray, LPCI, and HPCI systems are filled provides for a visual observation that water flows from a high point vent. This ensures that the line is in a full condition. Between the monthly intervals at which the lines are vented, instrumentation has been provided to monitor the presence of water in the discharge piping. This instrumentation will be calibrated on the same frequency as the safety system instrumentation. This period of periodic testing ensures that during the intervals between the monthly checks the status of the discharge piping is monitored on a continuous basis.

I. Average Planar LHGR

At core thermal power levels less than or equal to 25 per cent, operating plant experience and thermal hydraulic analyses indicate that the resulting average planar LHGR is below the maximum average planar LHGR by a considerable

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4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

margin; therefore, evaluation of the average planar LHGR below this power level is not necessary. The daily requirement for calculating average planar LHGR above 25 per cent rated thermal power is sufficient since power distribution shifts are slow when there have not been significant power or control rod changes.

J. Local LHGR

The LHGR for all fuel shall be checked daily during reactor operation at greater than or equal to 25 per cent power to determine if fuel burnup or control rod movement has caused changes in power distribution. A limiting LHGR value is precluded by a considerable margin when employing a permissible control rod pattern below 25% rated thermal power.

K. Minimum Critical Power Ratio (MCPR)

At core thermal power levels less than or equal to 25 per cent, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience and thermal hydraulic analysis indicates that the resulting MCPR value is in excess of requirements by a considerable margin. With this low void content, any inadvertent core flow increase would only place operation in a more conservative mode relative to MCPR.

The daily requirement for calculating MCPR above 25 percent rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

In addition, the reduced flow correction applied to the LCO provides margin for flow increase from low flows.

L. Flood Protection

The watertight bulkhead door and the penetration seals for pipes and cables penetrating the vault walls have been designed to withstand the maximum flood conditions. To assure that their installation is adequate for maximum flood conditions, a method of testing each seal has been devised.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

To test a pipe seal, another test seal is installed in the opposite side of the penetration creating a space between the two seals that can be pressurized. Compressed air is then supplied to a fitting on the test seal and the space inside the sleeve is pressurized to approximately 15 psi. The outer face of the permanent seal is then tested for leaks using a soap bubble solution.

On completion of the test, the test seal is removed for use on other pipes and penetrations of the same size.

In order to test the watertight bulkhead doors, a test frame must be installed around each door. At the time of the test, a reinforced steel box with rubber gasketing is clamped to the wall around the door. The fixture is then pressurized to approximately 15 psig to test for leak tightness.

Floor drainage of each vault is accomplished through a carbon steel pipe which penetrates the vault. When open, this pipe will drain the vault floor to a floor drain sump in the condensate pump room.

Equipment drainage from the vault coolers and the CCSW pump bedplates will also be routed to the vault floor drains. The old equipment drain pipes will be permanently capped to preclude the possibility of back-flooding the vault.

As a means of preventing backflow from outside the vaults in the event of a flood, a check valve and an air operated valve are installed in the 2" vault floor drain line 6'0" above the floor of the condensate pump room.

The check valve is a 2" swing check designed for 125 psig service. The air operated valve is a control valve designed for a 50 psi differential pressure. The control valve will be in the normally open position in the energized condition and will close upon any one of the following:

a. Loss of air or power

b. High level (5'0") in the condensate pump room

Closure of the air operated valve on high water level in the condensate pump room is effected by use of a level switch set at a water level of 5'0". Upon actuation, the switch will close the control valve and alarm in the control room.

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

The operator will also be aware of problems in the vaults/ condensate pump room if the high level alarm on the equipment drain sump is not terminated in a reasonable amount of time. It must be pointed out that these alarms provide information to the operator but that operator action upon the above alarms is not a necessity for reactor safety since the other provisions provide adequate protection.

A system of level switches has been installed in the condenser pit to indicate and control flooding of the condenser area. The following switches are installed:

	Level	Function
a.	1'0" (1 switch)	Alarm, Panel Hi-Water-Condenser Pit
b.	3'0" (1 switch)	Alarm, Panel High-Circ. Water Condenser Pit
c.	5'0" (2 redundant switch pairs)	Alarm and Circ. Water Pump Trip

Level (a) indicates water in the condenser pit from either the hotwell or the circulating water system. Level (b) is above the hotwell capacity and indicates a probable circulating water failure.

Should the switches at level (a) and (b) fail or the operator fail to trip the circulating water pumps on alarm at level (b), the actuation of either level switch pair at level (c) shall trip the circulating water pumps automatically and alarm in the control room. These redundant level switch pairs at level (c) are designed and installed to IEEE-279, "Criteria for Nuclear Power Plant Protection Systems." As the circulating water pumps are tripped, either manually or automatically, at level (c) of 5'0", the maximum water level reached in the condenser pit due to pumping will be at the 491'0" elevation (10' above condenser pit floor elevation 481'0"; 5' plus an additional 5' attributed to pump coastdown).

In order to prevent overheating of the CCSW pump motors, a vault cooler is supplied for each pump. Each vault cooler is designed to maintain the vault at a maximum 105°F temperature during operation of its respective pump. For example, if CCSW pump 2B-1501 starts, its cooler will also start and compensate

4.5 SURVEILLANCE REQUIREMENT BASES (Cont'd.)

for the heat supplied to the vault by the 2B pump motor keeping the vault at less than 105°F.

Each of the coolers is supplied with cooling water from its respective pump's discharge line. After the water has been passed through the cooler, it returns to its respective pump's suction line. In this way, the vault coolers are supplied with cooling water totally inside the vault. The cooling water quantity needed for each cooler is approximately 1% to 5% of the design flow of the pumps so that the recirculation of this small amount of heated water will not affect pump or cooler operation.

Operation of the fans and coolers is required during pump operability testing and thus additional surveillance is not required.

Verification that access doors to each vault are closed, following entrance by personnel, is covered by station operating procedures.

ATTACHMENT 2

DISCUSSION OF SUPPLEMENTAL DRESDEN 2

TECHNICAL SPECIFICATION CHANGES

As part of EPRI's Alternate Water Chemistry program (also referred to as Hydrogen Addition), General Electric (GE) is assessing its effects on fuel rod performance. As was done at the end of Cycle 9, three fuel rods are being removed from a precharacterized GE Lead Test Assembly (LTA) for shipment to GE for destructive examination. Rods are being taken from a higher burnup GE assembly (LJU345) to replace the three removed rods. Due to the lack of perfectly matched replacement rods, the reconstituted assembly (bundle LY5458) will have a slightly increased local peaking factor (approximately 3%).

To protect applicable GE fuel design criteria, as outlined in GESTAR, a small reduction in the MAPLHGR curve for this fuel assembly is required. It should be noted that although MAPLHGR curves were originally established to assure compliance with Emergency Core Cooling System (ECCS) analyses, GE also uses them to assure their fuel assemblies meet other applicable Thermal, Hydraulic, and Mechanical design criteria. Although the reconstituted fuel assembly meets all ECCS criteria without a MAPLHGR change, other design criteria (e.g., centerline melt, cladding strain, etc.) are not met without the change. GE's evaluation of the bundle reconstitution which includes the revised MAPLHGR limits for this bundle is provided in Attachment 3.

During a teleconference on January 20, 1987, the NRC Staff expressed concern over the use of the general fuel design terms "8x8" and "9x9" which are found in Section 3.5 of the Technical Specifications. Attachment 1 includes a clarified use of "8x8" and "9x9" in describing fuel types by stating "GE and Exxon 8x8" and "Exxon 9x9".

Attachment 1 includes pages which requires new page numbers or new sheet number designations due to the addition of the MAPLHGR curve for GE LTA LY5458. The Table of Contents and List of Figures are also included for the same reasons. Since these changes supplement an earlier submittal which is still pending NRC acceptance, revision bars were drawn for changes made in both the earlier and this supplementary submittal.

ATTACHMENT 3

GENERAL ELECTRIC EVALUATION OF

BUNDLE RECONSTITUTION

2649K

GENERAL ELECTRIC

NUCLEAR ENERGY BUSINESS OPERATIONS
GENERAL ELECTRIC COMPANY • 175 CURTNER AVENUE • SAN JOSE, CALIFORNIA 95125
Mail Code 174

January 6, 1987
REP:87-008

cc: H. E. Bliss
R. Leong

Mr. J. L. Anderson
Fuel Buyer
Commonwealth Edison Company
Fuel Department, 234 E
P. O. Box 767
Chicago, IL 60690

Subject: DRESDEN ALTERNATE WATER CHEMISTRY LTA RECONSTITUTION

Attachment: Safety Analysis Report, "Dresden Hydrogen Water Chemistry LTA Reconstitution"

Dear Mr. Anderson:

The attached Safety Analysis Report is provided for your review.

Very truly yours,

R. E. Parr
Senior Fuel Project Manager
Edison Projects
M/C 174; (408) 925-6526

REP:gs

Attachment

Dresden Hydrogen Water Chemistry LTA Reconstitution

Background

As part of the ongoing effort at Dresden 2 to evaluate the long-term effects of hydrogen injection on fuel bundle cladding, it has been decided to remove three rods (one gadolinia rod and two UO_2 rods) from the Lead Test Assembly (LTA) bundle (LY5458) at the end of Cycle 10 for examination. To continue the surveillance program, the LTA will be reconstituted using three replacement rods from a GE bundle with similar mechanical and nuclear characteristics, and placed back into the core. Four bundles scheduled for discharge at the end of Cycle 10 (EOC10) were evaluated as candidates from which the donor rods could be selected. Fuel bundle LJU345 (a P8DRB265-6G3.0-80M-145 bundle loaded in Cycle 8) was selected because it had the lowest peak nodal exposure of the four candidates with a projected EOC10 exposure of 31,650 MWd/MTU. The following EOC11 exposures are projected if fuel rods from LJU345 are used:

	Exposure (MWd/MTU)	
	EOC10	EOC11
Bundle Avg - LY5458	19,204	28,602
Bundle Avg - LJU345	31,650	-
Replacement Rods*		
Rod Avg - G3	28,868	39,016
G6	32,001	41,536
F7	32,001	41,536
Peak Pellet - G3	-	45,532
G6	-	48,473
F7	-	48,473

* Replacement rods from Bundle LJU345 are inserted into the same rod locations in bundle LY5458 at the beginning of Cycle 11 (BOC11) (see Figure 1).

In addition to the replacement fuel rods, some of the LTA components will also be replaced.

Safety Analysis

In order to assure safe operation of the reconstituted bundle, the three areas related to bundle operation have been assessed. The results of this assessment are given below.

A. Fuel Rod Thermal-Mechanical Performance

The primary consideration is the performance of the high exposure replacement rods in the reconstituted fuel bundle. To assure that the reconstituted fuel bundle will operate within the conditions for which the fuel rods have been designed, MAPLHGRs as a function of exposure have been developed for the reconstituted fuel bundle. These MAPLHGRs reflect

the change in operating conditions for: a) the replacement rod containing gadolinia in position G3; b) the UO_2 replacement rods in positions G6 and G7; and c) the effect of the reconstitution on the remaining rods in the fuel bundle. The MAPLHGRs for the reconstituted fuel bundle are presented in Table 1.

Regarding the MAPLHGRs presented in Table 1, it should be noted that operation of the reconstituted fuel bundle to a peak planar average exposure of 31.5 GWd/STU will place it at the design and licensing basis peak pellet exposure of 50 GWd/MTU. Therefore, if operation of this bundle is planned beyond 31.5 GWd/STU, additional analyses are required.

B. Fuel Bundle Mechanical Compatibility and Performance

The LTA will contain the following new components (beyond the three new rods) once reconstituted:

a) 1 P8x8R lower tie plate with P8x8R non-barrier tie rod threads;

b) 2 P8x8R water rods (one spacer positioning);

c) 2 GE8x8E expansion springs located on the new water rods;

d) 7 P8x8R spacers;

e) 4 P8x8R finger springs (identical to original LTA finger springs); and

f) new nuts and lock tabs.

All other LTA components not listed above will be from the original LTA bundle.

The impact of these new components, including the three new fuel rods, has been reviewed with respect to fretting wear, fuel rod axial differential expansion, and compatibility with the LTA bundle. The conclusion is that the incorporation of the three new rods and the use of the new bundle hardware will not adversely impact the operability of the LTA bundle.

C. MCPR

The changes from reconstitution result in a negligible R-factor change, a slightly lower total bundle power, and a bundle reactivity which is lowered by approximately 0.5% delta k. From a nuclear standpoint, the net change in the reactivity due to the reconstitution of the fuel bundle is small enough that the reconstituted bundle can be modeled as if it is not reconstituted.

Conclusion

The changes made in the Dresden-2 Hydrogen Water Chemistry LTA as a result of its reconstitution do not adversely impact the capability of the bundle to safely operate for the duration of its scheduled lifetime. However, a change in the MAPLHGR Technical Specifications is required.

Table 1

MAPLHGR Versus Exposure for the Reconstituted Fuel Bundle (LY5458)

Planar Exposure (GWd/STU)	MAPLHGR (kW/ft)
15.0	11.8
20.0	11.6
25.0	10.9
30.0	10.2
35.0	9.6

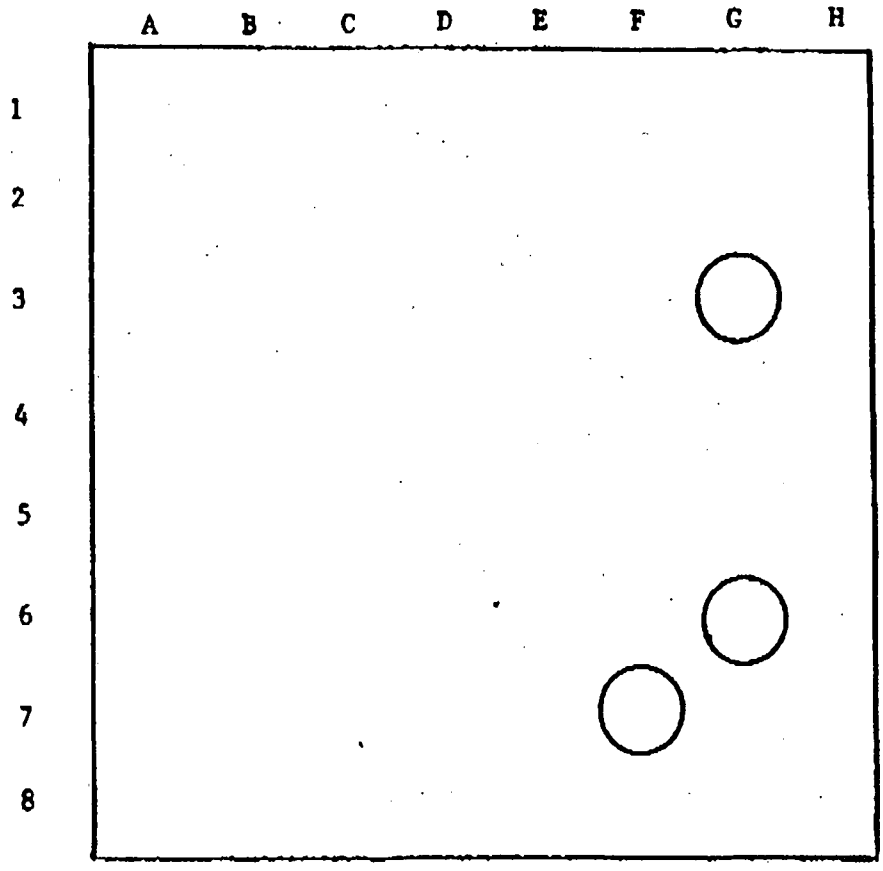

Figure 1

Location of Reconstituted Rods

ATTACHMENT 4

EVALUATION OF SIGNIFICANT HAZARDS CONSIDERATION

Description of Amendment Request

As part of EPRI's alternate water chemistry program, CECo and General Electric (GE) are assessing the effects of Hydrogen Water addition on fuel rod performance. Three fuel rods are being removed for a precharacterized GE Lead Test Assembly (LTA) for shipment to GE for destructive examination. Replacement rods are being taken from another irradiated GE assembly. As a result of reconstitution, the LTA will have an approximate 3% increase in local peaking; therefore, a revision to the applicable MAPLHGR curve is required to assure adequate protection of the fuel assembly design criteria.

CECo proposes to amend Provisional Operating License DPR-19 for Dresden Unit 2 to modify MAPLHGR curve for reconstituted GE assembly LY5458. In addition, minor labeling changes to specify fuel vendors have been included per NRC requests as well as pagination changes resulting from the MAPLHGR curve change. These changes are purely administrative and have no material affect on the operation of the plant. The impact of the proposed MAPLHGR change is discussed below.

Basis for Proposed No Significant Hazards Consideration Determination

CECo has evaluated the proposed Technical Specification amendment and determined that it does not represent a significant hazards consideration. Based on the criteria for defining the significant hazards consideration established in 10 CFR 50.92, operation of Dresden Unit 2 Cycle 11 with the reconstituted GE LTA will not:

1. Involve a significant increase in the probability or consequences of an accident previously evaluated;

   The reconstituted fuel assembly is mechanically identical to the original unreconstituted fuel assembly. Accident consequences are not increased by the small increase in local peaking in this assembly due the more restrictive MAPLHGR limits established by this amendment.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated because:

   All accidents involving reactor fuel, including refueling and fuel handling accidents, have previously been considered. The reconstituted assembly does not create any new scenarios beyond those previously analyzed.

3. Involve a significant reduction in the margin of safety because:

   The reconstituted fuel assembly has been reanalyzed by GE using NRC approved methodologies, as described in GESTAR; all applicable design criteria have been met. Therefore, the margin of safety is maintained.

Based on the above discussion, CECo concludes that the proposed amendments do not represent a significant hazards consideration.