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

May 7, 1987

Docket No. 50-331

Mr. Lee Liu
Chairman of the Board and
Chief Executive Officer
Iowa Electric Light and Power Company
Post Office Box 351
Cedar Rapids, Iowa 52406

Dear Mr. Liu:

SUBJECT: LICENSE AMENDMENT NO. 142- CYCLE 9 RELOAD (TAC 63568)

The Commission has issued the enclosed Amendment No. 142 to Facility Operating License No. DPR-49 for the Duane Arnold Energy Center. This amendment consists of changes to the Technical Specifications in response to your application dated October 31, 1986, and clarifying information dated March 20, 1987.

The amendment revises the Duane Arnold Energy Center (DAEC) Technical Specifications to support the reload and restart for Cycle 9 operation. The Technical Specification changes update the fuel thermal limits, revise the Limiting Conditions for Operation and Surveillance Requirements for the Rod Sequence Control System and Rod Worth Minimizer, and modify the description of the control blades.

A copy of the Safety Evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register Notice.

Sincerely,

for

Anthony J. Cappucci, Project Manager
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 142 to
License No. DPR-49
2. Safety Evaluation

cc w/enclosures:
See next page

Distribution and Concurrence for Letter to L. Liu dated - 5/7/87

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| RINGRAM | TCAPPUCCI:dlm | WHODGES | for MJVIRGILIO | |
| 4/30/87 | 4/30/87 | 4/30/87 | 5/1/87 | / /87 |

Conc. with SER as revised 5/5 (Redacted on Notice)

Mr. Lee Liu
Iowa Electric Light and Power Company

Duane Arnold Energy Center

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

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DOCKET NO. 50-331

DUANE ARNOLD ENERGY CENTER

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.142
License No. DPR-49

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

   A. The application for amendment by Iowa Electric Light and Power Company, et al., dated October 31, 1986, as clarified March 20, 1987, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

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

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C.(2) of Facility Operating License No. DPR-49 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 142, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. The license amendment is effective as of the date of issuance and shall be implemented within 30 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Martin J. Virgilio, Acting Director
Project Directorate III-1
Division of Reactor Projects - III, IV, V
& Special Projects

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 7, 1987

ATTACHMENT TO LICENSE AMENDMENT NO. 142

FACILITY OPERATING LICENSE NO. DPR-49

DOCKET NO. 50-331

Replace the following pages of the Appendix A Technical Specifications with the enclosed pages. The revised areas are indicated by marginal lines.

Pages

| | |
|---|---|
| vii | 3.12-3a |
| viii | 3.12-4 |
| 1.2-4 | 3.12-5 |
| 3.3-2 | 3.12-5a |
| 3.3-4 | 3.12-6 |
| 3.3-5 | 3.12-7 |
| 3.3-11 | 3.12-8 |
| 3.3-11a* | 3.12-9 |
| 3.3-14 | 3.12-10 |
| 3.3-15 | 3.12-12 |
| 3.3-16 | 3.12-13 |
| 3.3-17 | 3.12-14 |
| 3.3-20 | 3.12-17 |
| 3.5-26 | 3.12-20 |
| 3.12-1 | 3.12-22 |
| 3.12-2 | 3.12-23* |
| 3.12-3 | 5.2-1 |

*These pages are being deleted

# TECHNICAL SPECIFICATIONS

## LIST OF FIGURES

design pressure (120% x 1150 = 1380 psig; 120% x 1325 = 1590 psig).

The analysis of the worst overpressure transient, a 3 second closure of all main steam isolation valves with a direct valve position scram failure (i.e., scram is assumed to occur on high neutron flux), shows that the peak vessel pressure experienced is much less than the code allowable overpressure limit of 1375 psig (Reference 1). Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected overpressure transients.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. At this time it is included in the reactor coolant system.

## 1.2 References

1. Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center, Unit 1.*

*Refer to analyses for the current operating cycle.

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| b. The control rod directional control valves for inoperable control rods shall be disarmed electrically and the control rods shall be in such position that Specification 3.3.A.1 is met. | b. (DELETED) |
| c. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable. | c. Once per week when the plant is in operation, check status of pressure and level alarms for each CRD accumulator. |
| d. (DELETED) | d. Once per quarter verify that: (1) the Scram Discharge Volume (SDV) vent and drain valves close within 30 seconds after receipt of a close signal, and (2) after removal of the close signal, that the SDV vent and drain valves are open. Once per month verify that the SDV vent and drain valve position indicating lights located in the control room indicate that the valves are open. |
| e. Control rods with scram times greater than those permitted by Specification 3.3.C.3 are inoperable, but if they can be inserted with control rod drive pressure they need not be disarmed electrically. | e. Once per cycle verify that: (1) the SDV vent and drain valves close within 30 seconds after receipt of a signal for the control rods to scram, and (2) open when the scram signal is reset. |
| f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met. | |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | c. During each REFUELING OUTAGE observe that any drive which has been uncoupled from and subsequently recoupled to its control rod does not go to the overtravel position. |
| 2. The control rod drive housing support system shall be in place during REACTOR POWER OPERATION or 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. | 2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded. |
| 3.a Whenever the reactor is in the STARTUP or RUN mode below 30% RATED POWER, and the control rod movement is within the group notch mode after 50% of the control rods have been withdrawn, the Rod Sequence Control System (RSCS) shall be OPERABLE. If the system is determined to be inoperable in accordance with checks in Specification 4.3.B.3, power may be increased above 30% RATED POWER by increasing core flow. | 3.a. Prior to the start of control rod withdrawal towards criticality and prior to attaining 30% RATED POWER during rod insertion at shutdown, the capability of the Rod Sequence Control System to properly fulfill its function shall be verified by the following check:<br><br>Group Notch - Test the six comparator circuits. Go through each comparator inhibit, initiate test, verify error, and reset. After comparator checks initiate test and observe completion of cycle indicated by illumination of test complete light. |
| b. Whenever the reactor is in the STARTUP or RUN modes below 30% RATED POWER the Rod Worth Minimizer (RWM) shall be OPERABLE or a second Reactor Operator shall verify that the Reactor Operator at the reactor console is following the control rod program. | b. Prior to the start of control rod withdrawal towards criticality and prior to attaining 30% RATED POWER during rod insertion at shutdown, the capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:<br><br>1) The correctness of the Reduced Notch Worth Procedure sequence input to the RWM computer shall be verified. |
| c. If either Specifications 3.3.B.3.a or .b cannot be met, the reactor shall not be started, or if the reactor is in the RUN or STARTUP modes at less than 30% RATED POWER, control rod movement shall not be permitted, except by a scram. Limited control rod movement is permitted for the purpose of determining RSCS or RWM OPERABILITY and shall be verified by a second Reactor Operator. | |

| LIMITING CONDITION FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| | 2) The RWM computer on line diagnostic test shall be successfully performed. |
| | 3) Proper annunciation of the selection error of at least one out-of-sequence control rod in each fully inserted group shall be verified. |
| | 4) 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. Control rods shall not be withdrawn in STARTUP or REFUEL modes unless at least two Source Range Monitor Channels have an observed count rate equal to or greater than three counts per second. | 4. Prior to control rod withdrawal in STARTUP or REFUEL modes, verify that at least two Source Range Monitor Channels have an observed count rate of at least three counts per second. |
| 5. During operation with Limiting Control Rod Patterns, either: <br> a. Both RBM channels shall be OPERABLE, or <br> b. With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, unless OPERABILITY is restored within this time period, or <br> c. With both RBM channels inoperable, control rod withdrawal shall be blocked until OPERABILITY of at least one channel is restored. | 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). |

maximum contribution to shut-down reactivity. If it is disarmed electrically in a non-fully inserted position, that position shall be consistent with the shutdown reactivity limitation stated in Specification 3.3.A.1. This assures that the core can be shut down at all times with the remaining control rods assuming the strongest operable control rod does not insert. Inoperable bypassed rods will be limited within any group to not more than one control rod of a (5 x 5) twenty-five control rod array. If damage within the control rod drive mechanism and, in particular, cracks in drive internal housings cannot be ruled out, then a generic problem affecting a number of drives cannot be ruled out. Circumferential cracks resulting from stress assisted intergranular corrosion have occurred in the collet housing of drives at several BWR's. This type of cracking could occur in a number of drives and if the cracks propagated until severance of the collet housing occurred, scram could be prevented in the affected rods. Limiting the period of operation with a potentially severed collet housing and requiring increased surveillance after detecting one stuck rod will assure that the reactor will not be operated with a large number of rods with failed collet housings.

At power levels below 20% of rated, abnormal control rod patterns could produce rod worths high enough to be of concern relative to the 280 calories per gram rod drop limit. In this range the RWM and the RSCS constrain the control rod patterns to those which involve only acceptable rod worths.

The Reduced Notch Worth Procedure for control rod withdrawal allows the Group Notch RSCS plants to take advantage of the Banked Position Withdrawal Sequence (BPWS) (Ref. 1). The BPWS has the advantage of having been proven statistically to have such low individual control rod worths that the possibility of a control rod drop accident (CRDA), which exceeds the 280 cal/gm peak fuel enthalpy limit, is precluded (Ref. 2).

The Rod Worth Minimizer and the Rod Sequence Control System provide automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. They serve as a backup to procedural control of control rod sequences, which limit the maximum reactivity worth of control rods. In the event that the Rod Worth Minimizer is out of service, when required, a second Reactor Operator or other qualified technical plant employee whose qualifications have been reviewed by the NRC can manually fulfill the control rod pattern conformance functions of this system. In this case, the RSCS is backed up by independent procedural controls to assure conformance.

The functions of the RWM and RSCS make it unnecessary to specify a license limit on rod worth to preclude unacceptable consequences in the event of a CRDA. At low powers, below 20%, these devices force adherence to acceptable rod patterns. Above 20% of rated power, no constraint on rod pattern is required to assure that the consequences of a CRDA are acceptable.

Power level for automatic cutout of the RSCS function is sensed by first stage turbine pressure. Because the instrument has an instrument error of ± 10% full power, the nominal instrument setting is 30% of rated power. Power level for automatic cutout of the RWM function is sensed by feedwater and steam flow and is set nominally at 30% of rated power to be consistent with the RSCS setting.

The Reduced Notch Worth Procedure is programmed into the RWM and is compatible with the hardwired Group Notch RSCS. In the pre-checkerboard pattern (100% to 50% control rod density), the RWM will enforce the Reduced Notch Worth Procedure; while in the post-checkerboard pattern (50% control rod density to RSCS/RWM low power setpoint) the RSCS will enforce the rod pattern. Therefore, the RSCS is not required to be OPERABLE until the post-checkerboard pattern is entered.

Functional testing of the RWM prior to the start of control rod withdrawal at startup, and prior to attaining 30% rated thermal power during rod insertion while shutting down, will ensure reliable operation and minimize the probability of the rod drop accident.

The RSCS can be functionally tested prior to control rod withdrawal for reactor startup. The hardware functional test sequence is performed to demonstrate that the Group Notch mode of the

RSCS is OPERABLE prior to entering the Group Notch mode (i.e., after 50% control rod density). The Group Notch restraints are automatically removed above 30% power.

During reactor shutdown, similar surveillance checks shall be made with regard to rod group availability as soon as automatic initiation of the RSCS occurs and subsequently at appropriate stages of the control rod insertion.

If the operability requirements of either the RSCS or RWM are not satisfied, i.e., RSCS is inoperable or RWM is inoperable without the second reactor operator, then further rod movement is not permitted, except by a scram (manual scram or mode switch to SHUTDOWN). This is done to ensure that high rod worths, with the potential to exceed 280 cal/gm during a CRDA are not generated. However, limited rod movement shall be permitted solely for the purpose of troubleshooting and/or testing the RSCS or RWM for OPERABILITY. Limited rod movement is defined as the movement of control rod(s) only to the extent necessary to determine that the rod inhibit functions of RSCS or RWM are working properly.

In addition, if the RSCS become inoperable and reactor power is less than 30% of rated, but feedwater flow is above the interlock at 20% of rated feedwater flow, reactor power may be increased above the RSCS low power setpoint (30% rated power) by increasing the core flow. Increasing the power, without moving control rods, will ensure that a potential CRDA will not exceed the 280 cal/gm limit mentioned earlier, absent the automatic rod pattern constraints of the RSCS.

d. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. The consequences of reactivity accidents are functions of the initial neutron flux. The requirement of at least 3 counts per second assures that any transient, should it occur, begins at or above the initial value of $10^{-8}$ of rated power used in the analyses of transients cold conditions. One operable SRM channel would be adequate to monitor the approach to criticality using homogeneous patterns of scattered control rod withdrawal. A minimum of two operable SRM's are provided as an added conservatism.

e. The RBM provides local protection of the core; i.e., the prevention of boiling transition in a local region of the core, for a single rod withdrawal error from a Limiting Control Rod Pattern. The trip point is referenced to power. This power signal is provided by the APRMs. A statistical analysis of many single control rod withdrawal errors has been performed and at the 95/95 level the results show that with the specified trip settings, rod withdrawal is blocked at MCPRs greater than the Safety Limit, thus allowing adequate margin. This analysis assumes a steady state MCPR of 1.20 prior to the postulated rod withdrawal error. The RBM functions are required when core thermal power is greater than 30% and a Limiting Control Rod Pattern exists. When both RBM channels are operating either channel will assure required withdrawal blocks occur even assuming a single failure of one channel. When a Limiting Control Rod Pattern exists, with one RBM channel inoperable for no more than 24 hours, testing of the RBM prior to withdrawal of control rods assures that improper control rod withdrawal will be blocked (Reference 3). Requiring at least half of the normal LPRM inputs to be operable assures that the RBM response will be adequate to protect against rod withdrawal errors, as shown by a statistical failure analysis.

The RBM bypass time delay is set low enough to assure minimum rod movement while upscale trips are bypassed.

A Limiting Control Rod Pattern for rod withdrawal error (RWE) exists when (a) core thermal power is greater than or equal to 30% of rated and less than 90% of rated ($30\% \leq P < 90\%$) and the MCPR is less than 1.70, or (b) core thermal power is greater than or equal to 90% of rated ($P \geq 90\%$) and the MCPR is less than 1.40.

## 3.3 and 4.3 REFERENCES


1) General Electric Service Information Letter (SIL) No. 316, Reduced Notch Worth Procedure, November 1979.

2) General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A*.

3) Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center, NEDC-30813-P, December, 1984.


*Latest NRC-approved revision.

## 3.5 REFERENCES


1. Jacobs, I.M., Guidelines for Determining Safe Test Intervals and Repair Times for Engineered Safeguards, General Electric Company, APED, April 1968 (APED 5736).

2. General Electric Company, The GESTR-LOCA and SAFER Models for the Evaluation of Loss-of-Coolant Accident, NEDC-23785-P, October 1984.

3. General Electric, Duane Arnold Energy Center SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, NEDC-31310-P, August 1986.

4. General Electric Company, Analysis of Reduced RHR Service Water Flow at the Duane Arnold Energy Center, NEDE-30051-P, January 1983.

5. General Electric Company, Duane Arnold Energy Center Suppression Pool Temperature Response, NEDC-22082-P, March 1982.

LIMITING CONDITIONS FOR OPERATION

## 3.12 CORE THERMAL LIMITS

### Applicability

The Limiting Conditions for Operation associated with the fuel rods apply to those parameters which monitor the fuel rod operating conditions.

### Objective

The Objective of the Limiting Conditions for Operation is to assure the performance of the fuel rods.

### Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

1. During REACTOR POWER OPERATION, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-4, -5, -6, -7, -8 and -9 multiplied by the smaller of the two MAPFAC factors determined from Figs. 3.12-10 and 3.12-11.

2. During SLO, the actual MAPLHGR for each type of fuel as a function of average planar exposure shall not exceed the limiting value shown in Figs. 3.12-4, -5, -6, -7, -8 and -9 multiplied by the smaller of the two MAPFAC factors determined from Figs. 3.12-11 and 3.12-12.

SURVEILLANCE REQUIREMENTS

## 4.12 CORE THERMAL LIMITS

### Applicability

The Surveillance Requirements apply to the parameters which monitor the fuel rod operating conditions.

### Objective

The Objective of the Surveillance Requirements is to specify the type and frequency of surveillance to be applied to the fuel rods.

### Specifications

A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

The MAPLHGR for each type of fuel as a function of average planar exposure shall be determined daily during reactor operation at $\geq$ 25% RATED POWER.

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 3. If at any time during REACTOR POWER OPERATION (one or two loop) at $\geq$ 25% RATED POWER, it is determined by normal surveillance that the limiting value for MAPLHGR (LAPLHGR) is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the MAPLHGR (LAPLHGR) is not returned to within the prescribed limits within 2 hours, reduce reactor power to $\leq$ 25% of RATED POWER, or to such a power level that the limits are again being met, within the next 4 hours. | |
| 4. If the reactor is being operated in SLO and cannot be returned to within prescribed limits within this 4 hour period, the reactor shall be brought to the COLD SHUTDOWN condition within 36 hours. | |
| 5. For either the one or two loop operating condition surveillance and corresponding action shall continue until the prescribed action is met. | |
| B. <u>Linear Heat Generation Rate (LHGR)</u> | B. <u>Linear Heat Generation Rate (LHGR)</u> |
| 1. During REACTOR POWER OPERATION the LHGR of any rod in any BP/P8X8R or ELTA fuel assembly shall not exceed 13.4 KW/ft, while the LHGR of any rod in any GE8X8EB or LTA 311 fuel assembly shall not exceed 14.4 KW/ft. | The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% RATED POWER. |

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| 2. If at any time during REACTOR POWER OPERATION at $\geq$ 25% RATED POWER it is determined by normal surveillance that the limiting value for LHGR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the LHGR is not returned to within the prescribed limits within 2 hours, reduce reactor power to $\leq$ 25% of RATED POWER, or to such a power level that the limits are again being met, within the next 4 hours. Surveillance and corresponding action shall continue until the prescribed limits are again being met. | |
| C. Minimum Critical Power Ratio (MCPR) | C. Minimum Critical Power Ratio (MCPR) |
| 1. During REACTOR POWER OPERATION, the MCPR shall be equal to or greater than the Operating Limit MCPR, which is a function of core thermal power, core flow, fuel type and scram time ($\tau$). For core thermal power greater than or equal to 25% of rated and less than 30% of rated (25% $\leq$ P < 30%), the Operating Limit MCPR is given by Fig. 3.12-2. For core thermal power greater than or equal to 30% of rated (P $\geq$ 30%), the Operating Limit MCPR is the greater of either:<br><br>a) The applicable flow-dependent MCPR ($MCPR_F$) determined from Figure 3.12-1, or<br><br>b) The appropriate RATED POWER MCPR from Figure 3.12-3 [MCPR(100)] multiplied by the applicable power-dependent MCPR multiplier | MCPR shall be determined daily during REACTOR POWER OPERATION at $\geq$ 25% RATED POWER and following any change in power level or distribution that would cause operation with a Limiting Control Rod Pattern as defined in Section 3.2.C.2(a). During operation with a Limiting Control Rod Pattern, the MCPR shall be determined at least once per 12 hours. |

| LIMITING CONDITIONS FOR OPERATION | SURVEILLANCE REQUIREMENT |
|---|---|
| ($K_p$) determined from Figure 3.12-2.<br><br>2. During SLO with core thermal power greater than or equal to 25% of rated, the Operating Limit MCPR is increased by adding 0.03 to the above determined Operating Limit MCPR.<br><br>3. If at any time during REACTOR POWER OPERATION (one or two recirc. loop) at $\geq$ 25% RATED POWER, it is determined by normal surveillance that the limiting value for MCPR is being exceeded, action shall then be initiated within 15 minutes to restore operation to within the prescribed limits. If the operating MCPR is not returned to within the prescribed limits within two hours, reduce reactor power to $\leq$ 25% of RATED POWER, or to such a power level that the limits are again being met, within the next 4 hours.<br><br>4. If the reactor is being operated in SLO, and cannot be returned to within prescribed limits within this 4 hour period, the reactor shall be brought to a COLD SHUTDOWN condition within 36 hours.<br><br>5. For either the one or two recirc. loop operating condition surveillance and corresponding action shall continue until the prescribed action is met. | |

## 3.12 BASES: CORE THERMAL LIMITS

### A. Maximum Average Planar Linear Heat Generation Rate (MAPLHGR)

This specification assures that the peak cladding temperature (PCT) following the postulated design basis Loss-of-Coolant Accident (LOCA) will not exceed the limits specified in 10CFR50.46 and that the fuel design analysis limits specified in NEDE-24011-P-A (Reference 1) will not be exceeded.

Mechanical Design Analysis: NRC approved methods (specified in Reference 1) are used to demonstrate that all fuel rods in a lattice operating at the bounding power history, meet the fuel design limits specified in Reference 1. No single fuel rod follows, or is capable of following, this bounding power history. This bounding power history is used as the basis for the fuel design analysis MAPLHGR limit.

LOCA Analysis: A LOCA analysis is performed in accordance with 10CFR50 Appendix K to demonstrate that the permissible planar power (MAPLHGR) limits comply with the ECCS limits specified in 10CFR50.46. The analysis is performed for the most limiting break size, break location, and single failure combination for the plant (Reference 2).

The Technical Specification MAPLHGR limit is the most limiting composite of the fuel mechanical design analysis MAPLHGR and the LOCA analysis MAPLHGR limit.

The actual MAPLHGR values for the GE 8 fuel design are lattice-type dependent and are explicitly modeled by the plant process computer. The lattice-type dependent values can be found in Reference 2. The Technical Specification MAPLHGR limit is a nominal representation of the lattice-dependent values, (i.e., the most limiting lattice-type, other than the natural uranium bundle ends), which can be used to conservatively model the MAPLHGR limit if the process computer becomes unavailable.

The flow-dependent correction factor (Figure 3.12-10) applied to the MAPLHGR limits at rated conditions assures that (1) the 10CFR50.46 limit would not be exceeded during a LOCA initiated from less than rated core flow conditions and (2) the fuel thermal-mechanical design criteria would be met during abnormal operating transients initiated from less than rated core flow conditions (Reference 5).

The power-dependent correction factor (Figure 3.12-11) applied to the MAPLHGR limits at rated conditions assures that the fuel thermal-mechanical design criteria would be met during abnormal operating transients initiated from less than rated power conditions (Reference 5).

For two recirculation loop operation, the calculational procedures used to establish the MAPLHGR's shown on Figures 3.12-4 thru 3.12-9 are documented in Reference 1. The reduction factors for SLO were derived in Reference 4.

B. <u>Linear Heat Generation Rate (LHGR)</u>

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation rate and that the fuel cladding 1% plastic diametral strain linear heat generation rate is not exceeded during any abnormal operating transient if fuel pellet densification is postulated. The LHGR as a function of core height shall be checked daily during reactor operation at $\geq$ 25% power to determine if fuel burnup, or control rod movement has caused changes in power distribution. For LHGR to be a limiting value below 25% rated thermal power, the Maximum Total Peaking Factor (MTPF) would have to be greater than 10 which is precluded by a considerable margin when employing any permissible control rod pattern.

C. Minimum Critical Power Ratio (MCPR)

1. Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.12.C are derived from the established fuel cladding integrity Safety Limit MCPR value, and an analysis of abnormal operational transients (Reference 1). For any abnormal operating transient analysis evaluation with the initial condition of the reactor being at the steady state operating limit it is required that the resulting MCPR does not decrease below the Safety Limit MCPR at any time during the transient assuming instrument trip settings given in Specification 2.1.

To assure that the fuel cladding integrity Safety Limit is not exceeded during any anticipated abnormal operational transient, the most limiting transients have been analyzed to determine which result in the largest reduction in critical power ratio (CPR). The type of transients evaluated were loss of flow, increase in pressure and power, positive reactivity insertion, and coolant temperature decrease.

The limiting transient, which determines the required steady state MCPR limit, is the transient which yields the largest ΔCPR. The minimum Operating Limit MCPR of Specification 3.12.C bounds the sum of the Safety Limit MCPR and the largest ΔCPR.

The required MCPRs at rated power [MCPR(100)] are determined using the GEMINI transient analysis methods described in Reference 1. These limits were derived by using the GE 67B scram times, given in Section 3.3.C, which are based upon extensive operating plant data, as well as GE test data. The ODYN Option B scram insertion times were statistically derived from the 67B data to ensure that the resulting Operating Limit from the transient analysis would, with 95% probability at the 95% confidence level, result in the Safety Limit MCPR not being exceeded. The scram time parameter ($\tau$), as calculated by the following formula, is a measure of the conformance of the actual plant control rod drive performance to that used in the ODYN Option-B licensing basis:

$$\tau = \frac{\tau_{ave} - \tau_b}{\tau_a - \tau_b}$$

where: $\tau_{ave}$ = average scram insertion time to Notch 38, as measured by surveillance testing

$\tau_b$ = scram insertion time to Notch 38 used in the ODYN Option-B Licensing Basis.

$\tau_a$ = 67B scram insertion time to Notch 38

As the average scram time measured by surveillance testing ($\tau_{ave}$), exceeds the ODYN Option B scram time ($\tau_b$), the MCPRs at rated power [MCPR(100)] must be adjusted using Figure 3.12-3.

2. MCPR Limits for Other Than Rated Power and/or Rated Flow Conditions

At less than 100% of rated power and/or flow the required Operating Limit MCPR is the larger value of the flow-dependent MCPR ($MCPR_F$) or the power-dependent multiplier ($K_p$) times the rated power MCPR [MCPR(100)] at the existing core power/flow state. The required Operating Limit MCPR is a function of flow in order to protect the fuel from inadvertent core flow increases such that the Safety Limit MCPR requirement can be assured.

The $MCPR_F$s were calculated such that, for the maximum core flow rate and core thermal power along a conservative load line, the limiting bundle's relative power was adjusted until the MCPR was slightly above the Safety Limit MCPR. Using this relative bundle power, the MCPRs were calculated at different points along this conservative load line corresponding to different core flows. The resulting $MCPR_F$s are given in Figure 3.12-1.

For operation above 30% of rated thermal power, the core power-dependent MCPR operating limit is the rated power MCPR [MCPR(100)], multiplied by the factor given in Figure 3.12-2, i.e., $K_p$. For operation below 30% of rated thermal power, where the direct scrams on turbine control valve fast closure and turbine stop valve closures are bypassed, absolute MCPR limits are established. This limit is taken directly from

Figure 3.12-2. This limit protects the fuel from abnormal operating transients, including localized events, such as a rod withdrawal error, other than those resulting from inadvertent core flow increases, which are covered by the flow-dependent MCPR limits. This power-dependent MCPR limit was developed based upon bounding analyses for the most limiting transient at the given core power level. Further information on the MCPR operating limits for off-rated conditions is presented in Reference 5.

At thermal power levels less than or equal to 25% of rated thermal power, operating plant experience indicates that the resulting MCPR value is in excess of the requirements by considerable margin. Therefore, monitoring of MCPRs below this power level is unnecessary. The daily monitoring of MCPRs above 25% of rated thermal power is sufficient, since power distribution shifts are very slow, provided that no significant changes in core flow or control rod pattern have taken place.

During SLO, the Operating Limit MCPR must be increased by 0.03 to account for the increased uncertainty in the core flow and Transversing In-core Probe (TIP) readings used in the statistical analyses to derive the Safety Limit MCPR (see Reference 4).

## 4.12 BASES: CORE THERMAL LIMITS

### C. Minimum Critical Power Ratio (MCPR) - Surveillance Requirement

At core thermal power levels less than or equal to 25%, 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 indicated 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 state relative to MCPR. The daily requirement for calculating MCPR above 25% rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes. The requirement for calculating MCPR when operating with a Limiting Control Rod Pattern assures that Safety Limit MCPR will not be violated given a single rod withdrawal error (Reference 5).

## 3.12 REFERENCES

1. General Electric Standard Application for Reactor Fuel, NEDE-24011-P-A*.

2. Duane Arnold Energy Center SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis, NEDC-31310-P, August 1986.

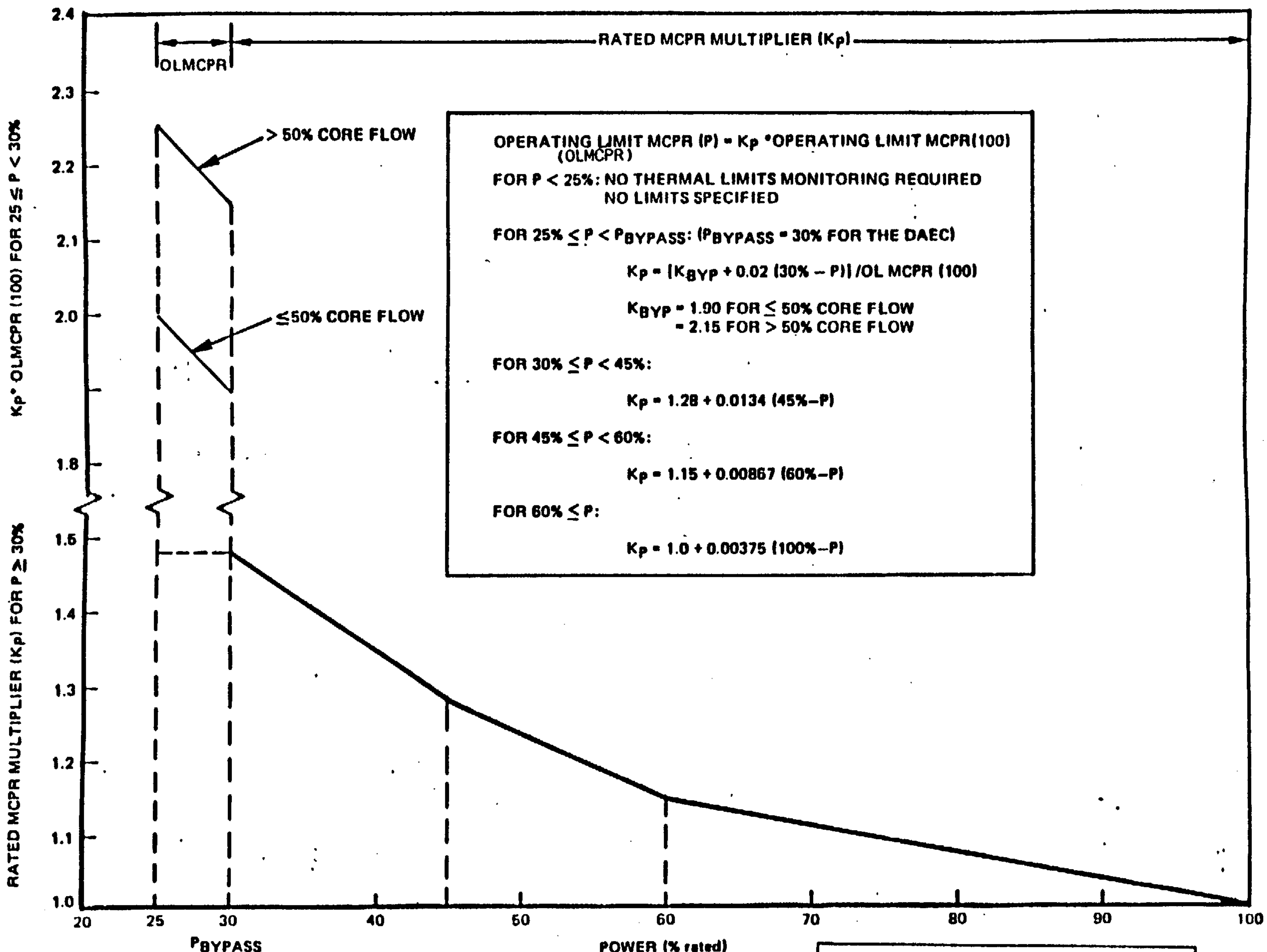
3. Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center, Unit 1.**

4. Duane Arnold Energy Center Single Loop Operation, NEDO-24272, July 1980.

5. Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) Program for the Duane Arnold Energy Center, NEDC-30813-P, December 1984.

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

**Analysis is cycle-dependent; see the report for the current operating cycle/reload.

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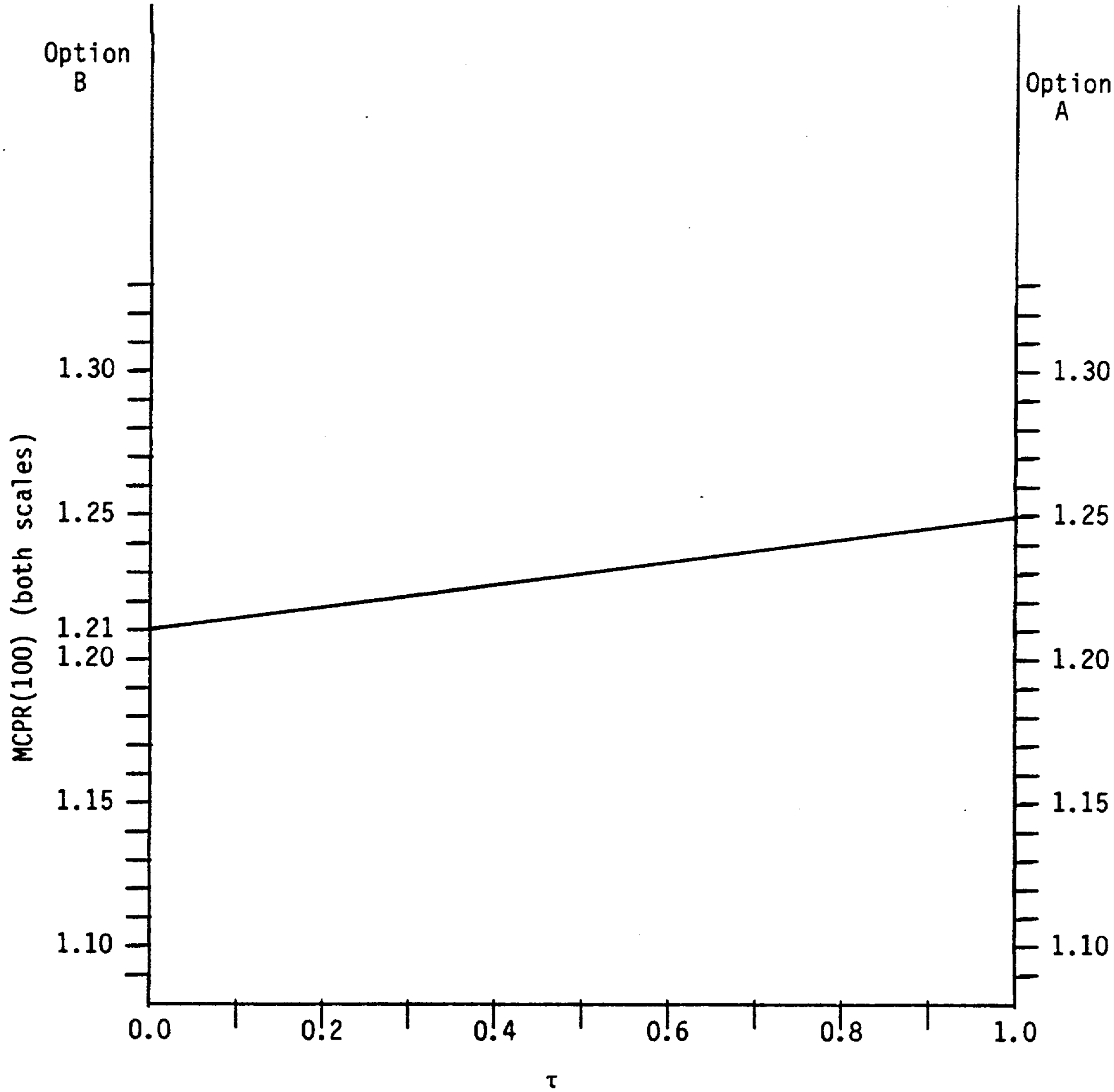
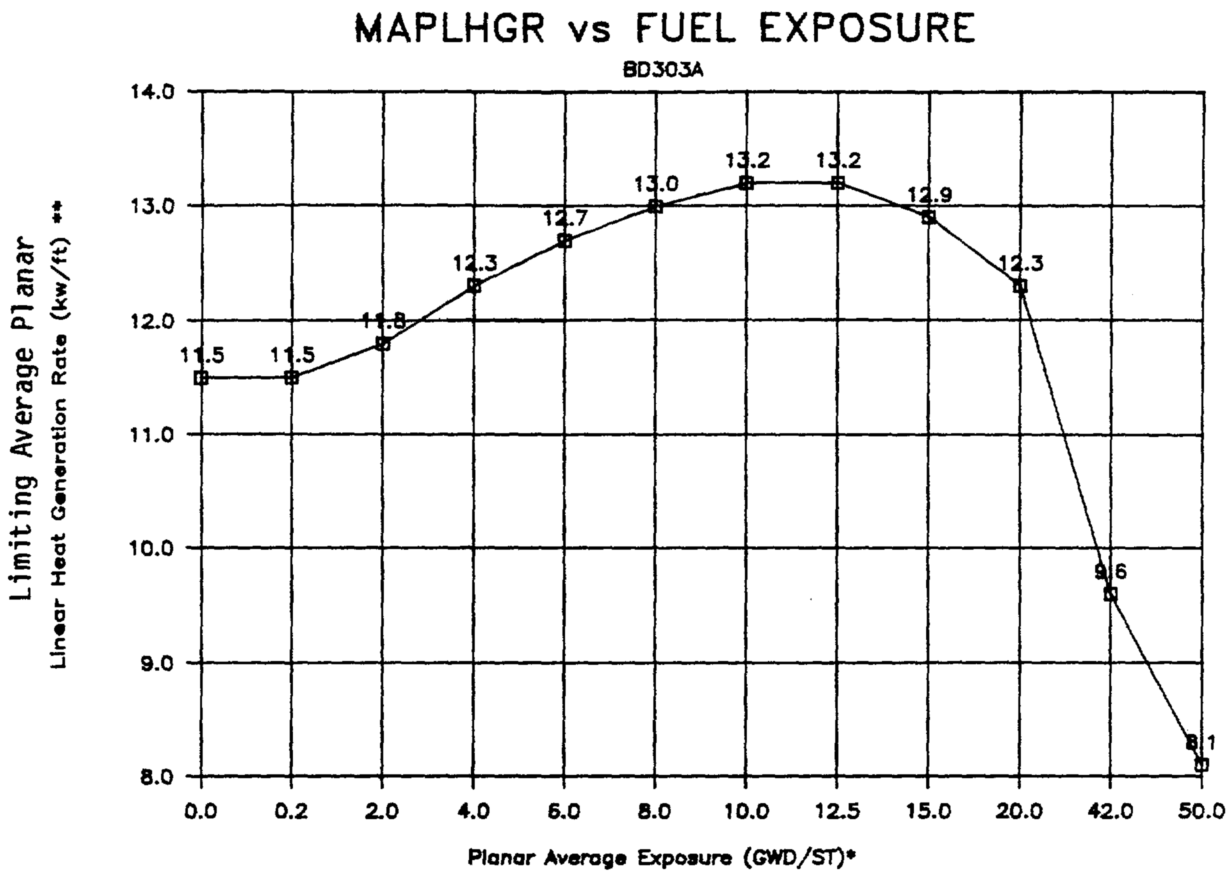
POWER-DEPENDENT MINIMUM CRITICAL POWER RATIO MULTIPLIER ($K_p$)
FIGURE 3.12-2




(based on tested measured scram time as defined in Reference 11)

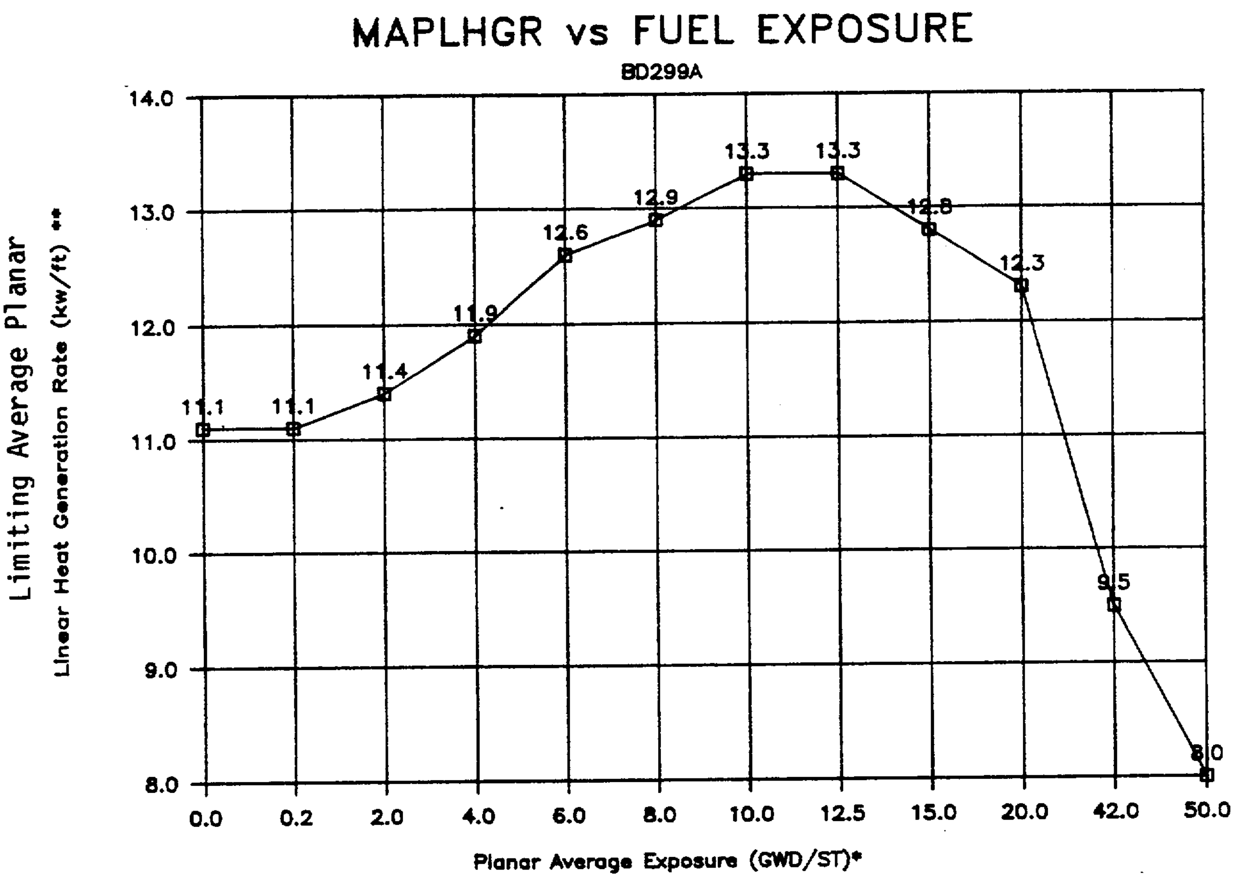
| |
|---|
| DUANE ARNOLD ENERGY CENTER<br>IOWA ELECTRIC LIGHT AND POWER COMPANY<br>TECHNICAL SPECIFICATIONS |
| MINIMUM CRITICAL POWER RATIO AT RATED POWER [MCPR(100)] VERSUS τ<br>FUEL TYPES: BP/P8X8R, GE8X8EB, LTA-311 and ELTA<br>FIGURE 3.12-3 |




* 1 GWd/t = 1000 MWd/t

** These values are nominal values to be used for manual calculations. The actual lattice-type dependent values are modeled in the process computer.

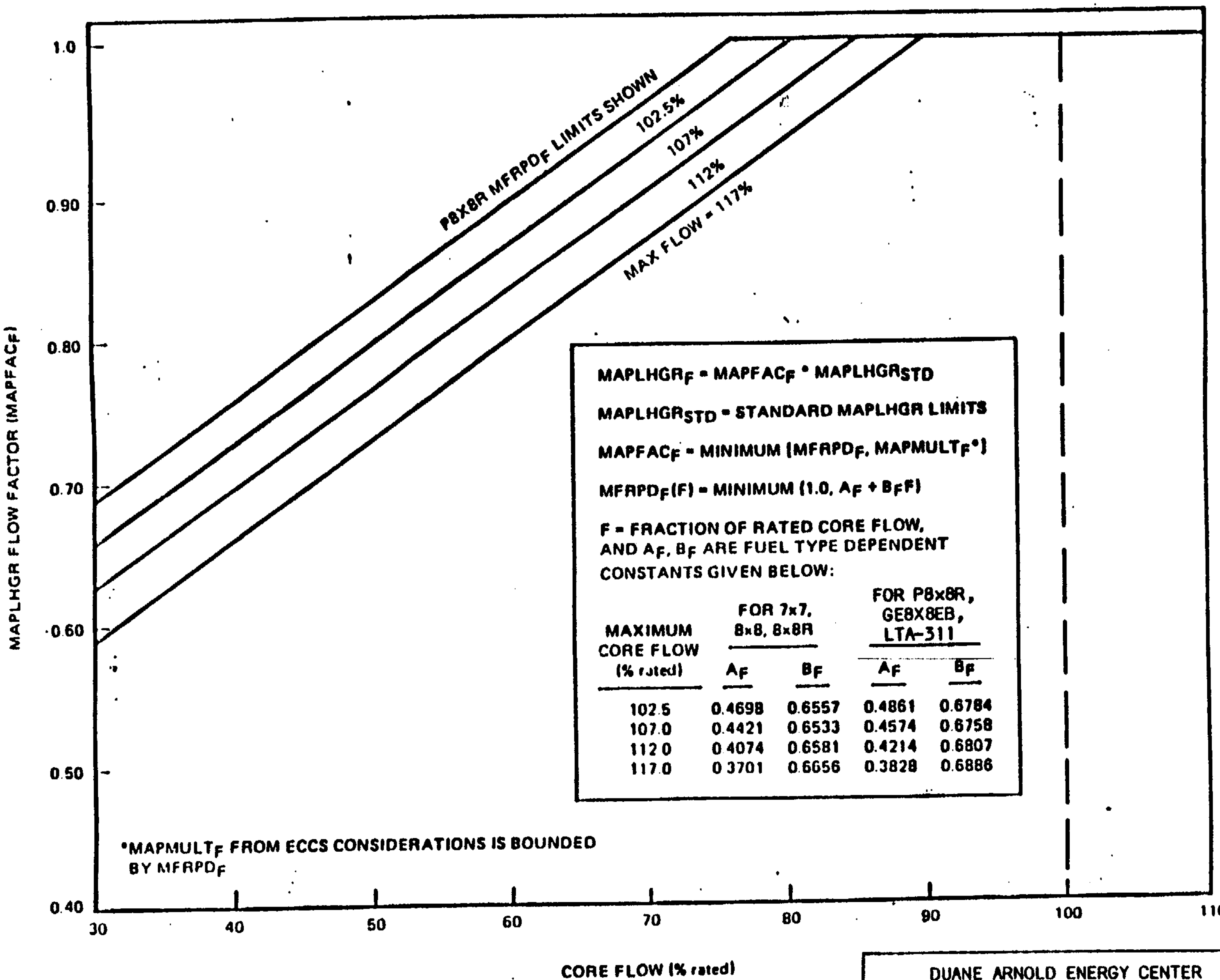
| DUANE ARNOLD ENERGY CENTER<br>IOWA ELECTRIC LIGHT AND POWER COMPANY<br>TECHNICAL SPECIFICATIONS |
|---|
| LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE<br>FUEL TYPE: BD303A<br>FIGURE 3.12-4 |




* 1 GWd/t = 1000 MWd/t

** These values are nominal values to be used for manual calculations. The actual lattice-type dependent values are modeled in the process computer.

| |
|---|
| DUANE ARNOLD ENERGY CENTER<br>IOWA ELECTRIC LIGHT AND POWER COMPANY<br>TECHNICAL SPECIFICATIONS |
| LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE<br>FUEL TYPE: BD299A<br>FIGURE 3.12-7 |




$MAPLHGR_F = MAPFAC_F * MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR LIMITS

$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$*)

$MFRPD_F(F)$ = MINIMUM (1.0, $A_F + B_F F$)

F = FRACTION OF RATED CORE FLOW, AND $A_F$, $B_F$ ARE FUEL TYPE DEPENDENT CONSTANTS GIVEN BELOW:

| MAXIMUM CORE FLOW (% rated) | FOR 7x7, 8x8, 8x8R | | FOR P8x8R, GE8X8EB, LTA-311 | |
|---|---|---|---|---|
| | $A_F$ | $B_F$ | $A_F$ | $B_F$ |
| 102.5 | 0.4698 | 0.6557 | 0.4861 | 0.6784 |
| 107.0 | 0.4421 | 0.6533 | 0.4574 | 0.6758 |
| 112.0 | 0.4074 | 0.6581 | 0.4214 | 0.6807 |
| 117.0 | 0.3701 | 0.6656 | 0.3828 | 0.6886 |

*$MAPMULT_F$ FROM ECCS CONSIDERATIONS IS BOUNDED BY $MFRPD_F$

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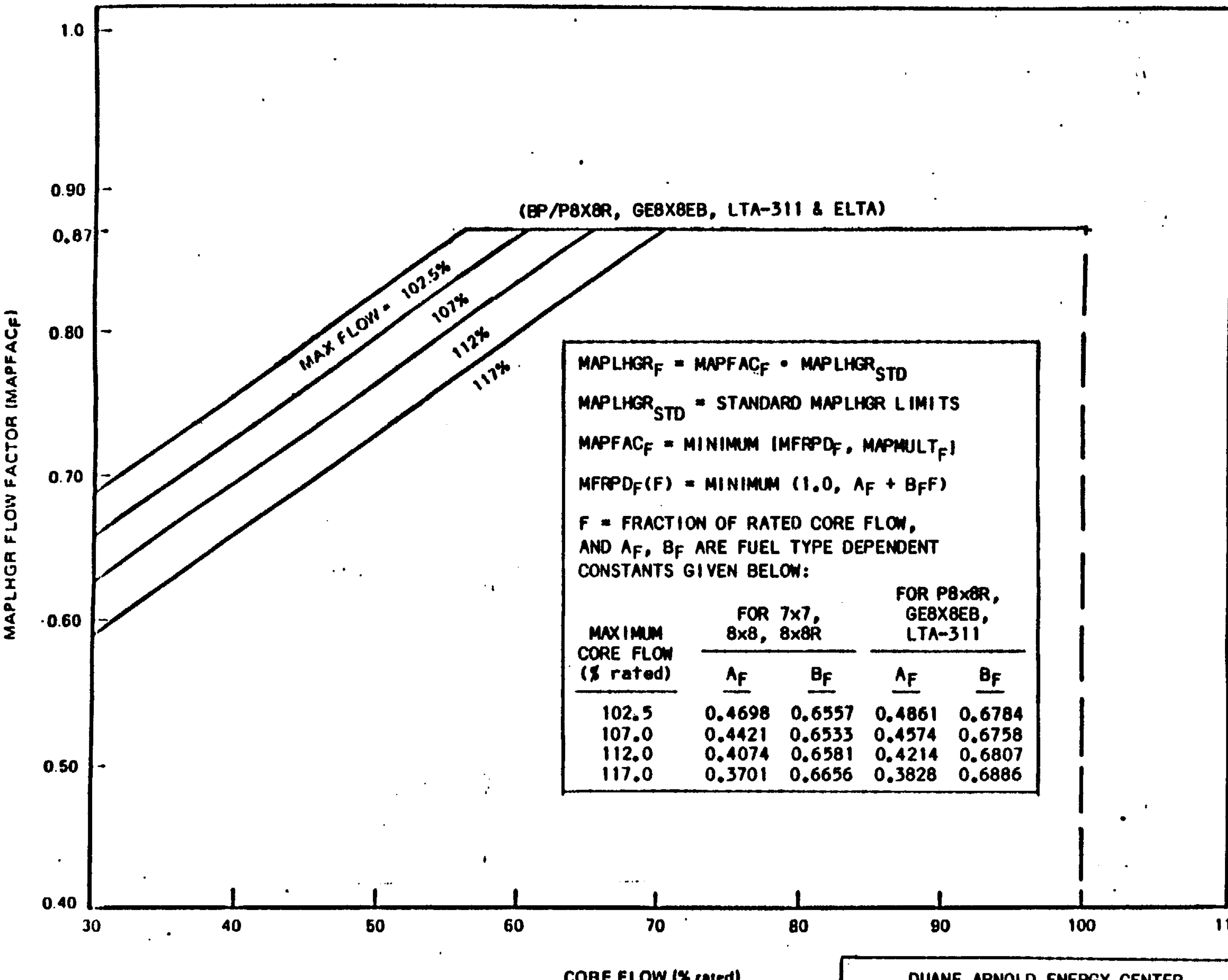
FLOW-DEPENDENT MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) MULTIPLIER ($MAPFAC_F$)
FIGURE 3.12-10




$MAPLHGR_F = MAPFAC_F \cdot MAPLHGR_{STD}$

$MAPLHGR_{STD}$ = STANDARD MAPLHGR LIMITS

$MAPFAC_F$ = MINIMUM ($MFRPD_F$, $MAPMULT_F$)

$MFRPD_F(F)$ = MINIMUM (1.0, $A_F + B_F F$)

F = FRACTION OF RATED CORE FLOW, AND $A_F$, $B_F$ ARE FUEL TYPE DEPENDENT CONSTANTS GIVEN BELOW:

| MAXIMUM CORE FLOW (% rated) | FOR 7x7, 8x8, 8x8R | | FOR P8x8R, GE8X8EB, LTA-311 | |
|---|---|---|---|---|
| | $A_F$ | $B_F$ | $A_F$ | $B_F$ |
| 102.5 | 0.4698 | 0.6557 | 0.4861 | 0.6784 |
| 107.0 | 0.4421 | 0.6533 | 0.4574 | 0.6758 |
| 112.0 | 0.4074 | 0.6581 | 0.4214 | 0.6807 |
| 117.0 | 0.3701 | 0.6656 | 0.3828 | 0.6886 |

DUANE ARNOLD ENERGY CENTER
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TECHNICAL SPECIFICATIONS

FLOW-DEPENDENT MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE (MAPLHGR) MULTIPLIER (MAPFAC$_F$) FOR SLO

## 5.2 REACTOR

1. The core shall consist of not more than 368 fuel assemblies of an approved fuel design.

2. The reactor core shall contain 89 cruciform shaped control rods of an approved design.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 142 TO FACILITY OPERATING LICENSE NO. DPR-49

IOWA ELECTRIC LIGHT AND POWER COMPANY
CENTRAL IOWA POWER COOPERATIVE
CORN BELT POWER COOPERATIVE

DUANE ARNOLD ENERGY CENTER

DOCKET NO. 50-331

## 1.0 INTRODUCTION

By a letter dated October 31, 1986, the Iowa Electric Light and Power Company (the licensee/IELP) submitted an application to amend the Duane Arnold Energy Center (DAEC) Technical Specifications (TSs). In response to staff questions, the licensee also submitted clarifying information in a letter dated March 20, 1987. The changes were proposed to support the DAEC reload and operation for Cycle 9, and to incorporate administrative changes reflecting revision to figure numbers, table of contents, references, and correction of errors.

During Cycle 9, the licensee proposes to utilize the latest General Electric fuel design (GE8B) and analytical methods for fuel analysis (SAFER/GSTR LOCA models and Gemini Physics). The GE8B fuel design and the improved analytical methods have been previously approved by the staff. The licensee also proposes to change the TSs by updating the fuel thermal limits of TS Section 3.12, revising the Limiting Conditions for Operation and Surveillance Requirements for the Rod Sequence Control System (RSCS) and Rod Worth Minimizer (RWM) in Sections 3.3.B.3 and 4.3.B.3 and modifying the Section 5.2 description of the control blades.

## 2.0 EVALUATION

The staff review of the licensee's October 31, 1986, submittal and subsequent March 20, 1987, clarifying information is summarized as follows:

### Fuel Mechanical Design

For Cycle 9, 128 irradiated fuel assemblies will be removed from the reactor core and replaced by General Electric 8x8E assemblies. The GE8x8E fuel is similar to that customarily used for BWR reloads and is described in Reference 3. The mechanical design methodology is described in Reference 5 and was used in this design for the GE8x8E fuel. Reference 5 has been approved by the staff in Reference 6 and its supplements.

Fuel Design

The new fuel for Cycle 8 is the GE extended burnup fuel GE8x8E. The fuel designations are BD299A and BD303A. This fuel type has been approved in the Safety Evaluation Report for Amendment 10 to GESTAR II (Ref. 6). The specific descriptions of this fuel have been submitted in Amendment 18 to GESTAR II, but since this amendment has not as yet been accepted, the fuel description has also been presented for DAEC in Reference 4. The fuel descriptions in Reference 4 are acceptable to the staff.

In operation, the GE8x8E fuel will be assigned a number of axial lattice regions and appropriate maximum average planar linear heat-generation rate (MAPLHGR) limits. The MAPLHGR limits have been determined by approved thermal-mechanical and loss-of-coolant accident (LOCA) analyses calculations and will be applied to each of these regions. There was extensive interaction between the staff, GE and the utility in determining an acceptable format for presentation of this information which is suitable for plant use and meeting staff requirements for TSs. References 7, 8 and 9 provide questions, responses and conclusions from these interactions. The process computer contains, and acts on, full details of the MAPLHGR information. The agreed upon TSs present the least and most limiting lattice MAPLHGR as a function of burnup. When hand calculations of MAPLHGR are required (process computer interactive), the most limiting values are used for all limits. These TSs are acceptable. A proprietary report, reviewed by the staff, available to the DAEC engineering staff, provides complete details of the lattice definitions and MAPLHGR limits.

The proposed linear heat generation rate (LHGR) limit for the GE8x8E fuel is 14.4 kW/ft (rather than the 13.4 for other GE fuel). This LHGR limit has been reviewed and accepted for this fuel in the GE extended burnup fuel review (Reference 10). This LHGR limit is acceptable for DAEC Cycle 9 operation.

Nuclear Design

The nuclear design and analysis of the Cycle 9 reload was performed with methods and techniques which are described in Reference 5. The results of the analyses are given in References 3 and 4. The results of the Duane Arnold analyses are within the range of those reload cores previously reviewed by the staff and found to be acceptable. We therefore conclude that the nuclear design and analysis of the Cycle 9 reload is acceptable.

Thermal-Hydraulic Design

The methods and procedures employed in the thermal-hydraulic (T-H) design and analysis of the Cycle 9 core are described in Reference 5. The value of 1.07 for the safety limit minimum critical power ratio (MCPR), approved in that reference, is used for Cycle 9. The methods and procedures used to obtain the operating limit MCPR are those described in Reference 5, approved in Reference 6 and are acceptable.

Thermal-Hydraulic Stability

The issue regarding thermal-hydraulic stability has been resolved during the staff's review of one loop operation (Ref. 11). The licensee has changed the TSs which provide operating limits and surveillance requirements for thermal-hydraulic stability. As a result of its review, the staff has determined that the revised TSs implement the recommendations of GE SIL-380 and are acceptable for both one and two loop operation.

Loss-of-Coolant Accident Analyses

The LOCA analyses were performed using the SAFER/GESTR code and the application methodology described in Reference 12. In Reference 15, the staff has specified the necessary conditions for demonstrating applicability of the SAFER/GESTR methodology. These conditions are:

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

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

3. Confirmation that the plant specific emergency core cooling system (ECCS) configuration is consistent with the referenced plant class ECCS configuration.

The licensee has reported the results of those analyses (Ref. 4) which are required to meet these conditions. Specifically, the analyses include break sizes from 0.1 $ft^2$ to 2.52 $ft^2$ (DBA recirculation suction line break). Seven different break sizes were analyzed in conjunction with ECCS failure combinations. A total of 16 cases were evaluated to establish the trend of PCT curves (nominal and Appendix K) versus break size.

The input parameters for both the nominal and Appendix K cases are within those used in the approved generic analyses. The ECCS configuration of Duane Arnold (4 LPCI, 2CS and 1 HPCI) is consistent with the ECCS configuration of a generic BWR 4. The results show that the design basis accident (DBA) recirculation suction line break with battery failure is the limiting case. The calculated PCT is 1036° F when nominal input values are used and 1565° F when Appendix K input values are used. The input parameters, the ECCS combination and the cases analyzed to establish the trend of PCT verse break size meet the staff requirements given above. The accident analyses have been performed using approved methods and the results meet the staff's acceptance criteria, therefore, these analyses are acceptable.

MCPR and MAPLHGR Limits

A safety limit MCPR has been imposed to assure that 99.9 percent of the fuel rods in the core will not experience boiling transition during normal operation and anticipated operational transients. As stated previously, the safety limit of 1.07 was used for Cycle 8.

To assure that the fuel cladding integrity safety limit MCPR will not be violated during any anticipated transient, the most limiting events were reanalyzed for this reload (Ref. 3) to determine which events result in the largest reduction in critical power ratio (CPR). The operating limit MCPR was then established by adding the largest reduction factor in the CPR to the safety limit MCPR. Since acceptable methods (Ref. 5) have been used, we find the MCPR TS changes to be acceptable.

The MAPLHGR limits specified in the proposed TS changes are less than or equal to the bounding MAPLHGR used in the SAFER/GESTR-LOCA analysis (Ref. 4) and are, therefore, acceptable.

## 3.0 TECHNICAL SPECIFICATION CHANGES

### Reload of Cycle 9

The TS changes proposed by the licensee reflect the new fuel for Cycle 9. These changes include the LHGR limit, MCPR operating limit and the MAPLHGR curve for the GE8x8E fuel. These proposed changes are acceptable since they are based upon approved analytical methods as discussed above.

### Revisions to RSCS and RWM Operability Requirements

In Amendment 12 to Reference 5, General Electric proposed that Group Notch plants which elect to change to Banked Position Withdrawal Sequence (BPWS) supervised by RWM for the first 50 percent of withdrawal may take credit for the Control Rod Drop Accident (CRDA) statistical analysis approved by the staff for BPWS plants. This would result in these plants being able to delete CRDA analysis from reload analysis procedures. As a result of the review, the staff has concluded that the proposed amendment is acceptable. The staff has also taken the position that plants electing to change to BPWS must provide a submittal to the NRC indicating that the BPWS patterns will be enforced and that related TSs would be changed as required to so indicate.

In response to the staff position, the licensee submitted the proposed changes to the TS. The DAEC, which has a hard-wired Group Notch RSCS, will incorporate the Reduced Notch Worth Procedure (RNWP) into the RWM. The RNWP is a BPWS compatible but more restrictive sequence. The RNWP will reinforce the control rod withdrawal procedure in the range of highest control rod worth (100% to 50% control rod density). The existing Group Notch mode of RSCS will continue to reinforce the rod withdrawal procedure in the range from 50% rod density to the low power setpoint of approximately 30% rated power. Based on these control rod withdrawal procedures, the DAEC TS is changed to reflect these new procedures. The staff has reviewed the TS changes to the RSCS and RWM and finds these changes acceptable.

### Hybrid I Control Blades

The IELP proposed to use several new Hybrid I Control Blades for Cycle 9 operation. The Hybrid I Control Blades contain both hafnium and boron carbide as neutron absorber materials. The Hybrid I Control Blades were designed by General Electric and described in Reference 13. The staff has reviewed and approved the Hybrid I Control Blades (Ref. 14). Therefore, the proposed use of the Hybrid I Control Blades for the DAEC is acceptable.

## 4.0 ENVIRONMENTAL CONSIDERATIONS

This amendment changes a requirement with respect to 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.

## 5.0 CONCLUSIONS

Based on the review described above, we conclude that the Duane Arnold Energy Center may be loaded and operated for Cycle 9. This conclusion is based on the following:

1. The safety analyses have been performed by previously approved methods and procedures;

2. The Cycle 9 core meets all of the staff's acceptance criteria.

The staff also concludes that the associated changes to the TSs for Cycle 9 operation, RSCS and RWM operability requirements, and Hybrid I control blades are acceptable.

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.

## 6.0 REFERENCES

1. Letter, Richard W. McGaughy (Iowa Electric Light and Power Company) to Harold Denton (NRC), October 31, 1986.

2. Letter, Richard W. McGaughy (Iowa Electric Light and Power Company) to Harold Denton (NRC), March 20, 1987.

3. Supplemental Reload Licensing Submittal for Duane Arnold Atomic Energy Center, Unit 1, Reload 8, General Electric, 23A 4812, September 1986.

4. General Electric Company, NEDO-31310P, Duane Arnold Energy Center, "SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis," August 1986.

5. GESTAR II - "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A-6, July 1986.

6. Approval letter, D. G. Eisenhut (NRC) to R. Gridley (GE) dated May 12, 1978, and supplements thereto, forming Appendix C to Reference 5.

7. NEDE-24081-P, Supplement 1 (and Eratta Sheet No. 11), November 1986, "Loss-of-Coolant Accident Analysis for Peach Bottom 2," Revision 1.

8. Letter (and attachments) from J. Gallagher, Philadelphia Electric Co.), to D. Muller (NRC) dated March 24, 1987, "Peach Bottom 2 Reload 7."

9. Letter from J. Charnley (GE) to W. Hodges (NRC) dated March 4, 1987, "Recommended MAPLHGR Technical Specifications for Multiple Lattice Fuel Designs."

10. Letter (and attachments) from C. Thomas (NRC) to J. Charnley (GE) dated May 28, 1985, "Acceptance for Referencing of Licensing Topical Report NEDE-24011-P-B, Amendment 10."

11. Letter, A. Thadani (NRC) to L. Liu (Iowa Electric Light and Power Company), May 28, 1985.

12. NEDE-23785-1-PA, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident" Volume I, II and III, General Electric Company, June 1984.

13. NEDE-22290, "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly," September 1983.

14. Approval letter, Cecil O. Thomas (NRC) to J. F. Klapproth (GE), "Acceptance for Referencing of Licensing Topical Report NEDE-22290, Safety Evaluation of the General Electric Hybrid I Control Rod Assembly," August 1983.

15. Approval letter, Cecil O. Thomas (NRC) to J. F. Quirk (GE), "Acceptance for Referencing of Licensing Topical Reports NEDE-23785, Revision 1, Volume III (P), The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident."

Dated: May 7, 1987