

PROPOSED CHANGE RTS-211 TO THE
DUANE ARNOLD ENERGY CENTER
TECHNICAL SPECIFICATIONS

The holders of license DPR-49 for the Duane Arnold Energy Center propose to amend Appendix A (Technical Specifications) to said license by deleting current pages and replacing them with the attached, new pages. The List of Affected Pages is given below.

As new bundle types are added to the reactor core, the design basis Loss-of-Coolant Accident (LOCA) must be reanalyzed and the Technical Specifications updated to include the operating limits of these new bundle types. As part of the reload for Cycle 9, General Electric (GE) has reanalyzed the design basis event for the two new bundle types (BD303A & BD299A) being added to the core. This analysis utilizes GE's new LOCA analysis methodology, SAFER/GESTR. The results of this analysis are reported in the attached updated LOCA report. This change to the Technical Specifications is to add the Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) operating limits for the new bundle types and to revise the MAPLHGR correction factors for off-rated flow conditions (MAPFAC_F) to include all fuel types, for both dual and single recirculation loop operation, based on this analysis.

Also as part of the reload licensing process for Cycle 9, GE has reanalyzed the most limiting abnormal operating transients as specified in Table 15.0-1 in the updated Final Safety Analysis Report. This analysis was performed using GE's new GEMINI transient analysis methodology. The results of this analysis are reported in the attached supplement to the Reload Licensing Submittal. This change to the Technical Specifications is to revise the Minimum Critical Power Ratio (MCPR) operating limits based on this analysis for all fuel types.

The new fuel being added to the core is of the advanced GE8B design. The improved thermal-mechanical design of this fuel allows a higher Linear Heat Generation Rate (LHGR) limit to be utilized than for the previous fuel designs. This higher LHGR limit is conditional upon satisfactorily demonstrating that the GE8B fuel is not MAPLHGR-limited, i.e., the fuel is in compliance with 10 CFR 50.46 acceptance criteria at the higher LHGR value. As the above-mentioned LOCA analysis demonstrates this compliance, a higher Technical Specification LHGR limit is proposed for the GE8B fuel.

As stated in the SER for Amendment 12 to NEDE-24011-P-A (GE Standard Application for Reactor Fuel), the NRC will allow plants with Group Notch Rod Sequence Control Systems, which utilize Banked Position Withdrawal Sequence (BPWS)-compatible control rod withdrawal sequences in their Rod Worth Minimizer, to be exempt from having to analyze the control rod drop accident on a cycle-by-cycle basis. This exemption is conditional upon modifying the necessary Technical Specifications to require the use of BPWS-compatible withdrawal sequences during plant startup and shutdown. Such Technical Specification changes are enclosed to support eliminating the rod drop accident analysis for Cycle 9 operation of the DAEC. In addition, several other changes regarding the operability requirements of the Rod Sequence Control System (RSCS) and the Rod Worth Minimizer (RWM) are being proposed. These changes adopt NRC guidelines as outlined in the

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Standard Technical Specifications (STS). First, existing Specification 3.3.B.3.b is being deleted, as it serves no safety function, conflicts with other RSCS requirements (e.g. 3.3.B.3.a) and is not consistent with the guidance in the STS. The second proposed change will delete Specification 3.3.B.3.d, regarding allowable control rod withdrawal sequences, as compliance cannot be directly demonstrated using in-plant instrumentation. A third change is proposed which will revise the action statement under Specification 3.3.B.3.e to be less restrictive, while still satisfying the safety intent of the present requirement. The fourth proposed change will delete a surveillance requirement for the RWM. Specification 4.3.B.3.c requires that the physical presence of the second reactor operator verifying the conformance to the control rod pattern be verified. This surveillance is judged to be superfluous and is not required by the STS. Therefore, it is being deleted. The final proposed change will modify the present allowance in Specification 3.3.B.3.a which permits power to be increased above 30% of rated, by increasing the core flow, if the RSCS becomes inoperable during a power descent. The proposed change will permit the same allowance during a reactor startup as well.

Cycle 9 operation of the DAEC will utilize GE's Hybrid I control blades, which contain both hafnium and boron carbide as neutron absorber materials. The design of these control blades has been reviewed by the NRC and found acceptable for use in domestic BWRs (per NRC SER to NEDE-22290, "Safety Evaluation of the General Electric Hybrid I Control Rod Assembly"). The attached revisions to the Technical Specifications will delete the specific reference to the neutron absorber material to be used and will state that the control blades must be of an approved design.

In addition, various administrative changes are being made, such as revising figure numbers, updating the table of contents, capitalizing defined terms, correcting references, revising Bases and clarifying language to improve understanding.

The changes being made are as follows:

- 1) Revise the List of Figures (pages vii and viii) to delete the MAPLHGR curves for the old fuel (Fig. 3.12-7); add the MAPLHGR curves for the new fuel (new Figures 3.12-4 and 3.12-7); consolidate Figures 3.12-2 and -3 (MCPR vs. τ) into one Figure (Fig. 3.12-3) and renumber existing Figure 3.12-4 as 3.12-2; delete Figure 3.12-12 (MAPFAC_F for LTA-311) and renumber Figure 3.12-13 as 3.12-12.
- 2) Correct the title of Reference 1 to Bases Section 1.2.
- 3) Revise Sections 3.3 and 4.3 (Reactivity Control) as follows:
 - a) Revise the Operability and Surveillance Requirements (3.3.B.3 and 4.3.B.3, respectively) for the RSCS and RWM to utilize the Reduced Notch Worth Procedure, i.e., the RWM will enforce the rod sequence in the 100%-50% control rod density range, while the RSCS will enforce the sequence between 50% rod density and the low power setpoint (~30% rated power). Delete Specifications 3.3.A.2.d, 4.3.A.2.b, and 3.3.B.3.f, which allow rods to be bypassed in the RSCS under certain circumstances, as they are no longer needed after the above changes.

- b) Capitalize all terms used on the affected pages which are defined in Section 1.0.
 - c) Revise Specification 3.3.B.3.a to allow power to be increased above 30% power by increasing core flow when RSCS is inoperable during startup, as well as during shutdown.
 - d) Delete Specifications 3.3.B.3.b, 3.3.B.3.d and 4.3.B.3.c dealing with allowable rod movement during shutdown, allowable control rod withdrawal sequences and verification of the second reactor operator during RWM inoperability, respectively. Also, re-index the remaining sections due to these deletions.
 - e) Revise the title of licensed operator to reactor operator, per License Amendment No. 136.
 - f) Revise current Specification 3.3.B.3.e to prohibit rod movement rather than requiring a plant shutdown, if the operability requirements for either RSCS or RWM are not satisfied. Also, add an allowance for limited rod movement to allow operability testing of the RSCS or RWM.
 - g) Update the Bases section to reflect the above changes.
 - h) Revise the References to Sections 3.3 and 4.3 to reflect the above changes.
- 4) Revise the References to Section 3.5 to reflect the new LOCA analyses methods and results for the DAEC.
- 5) Revise Sections 3.12 and 4.12 (Core Thermal Limits) as follows:
- a) Revise the LCO statements to include the MAPLHGR limits for the new fuel types and add the new MAPLHGR vs. Exposure Curves for the new fuel.
 - b) Delete the reference to separate MAPFAC multipliers for the LTA-311 fuel and delete existing Fig. 3.12-12 (MAPFAC_F for LTA-311).
 - c) Revise the MAPFAC_F curves for dual and single loop operation (Figs. 3.12-10 and 3.12-12, respectively) to show they are applicable to all fuel types.
 - d) Revise the LCO statements for LHGR to reflect the higher limit for the new GE8B fuel.
 - e) Revise Figure 3.12-3 to reflect the new MCPR(100) limits for all fuel types during Cycle 9 operation and delete Figure 3.12-2.
 - f) Renumber existing Figure 3.12-4 (K_p) as Figure 3.12-2.
 - g) Capitalize all terms used in Sections 3.12 and 4.12 which are defined in Section 1.0.

- h) Revise the Bases section to reflect the above changes.
 - i) Update the list of References for Section 3.12.
- 6) Replace in Section 5.2 the reference to a specific neutron absorber material used in the control blades with a statement that the control blades shall be of an approved design.

LIST OF AFFECTED 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
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3.3-14	3.12-10
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*page deleted

TECHNICAL SPECIFICATIONS

LIST OF FIGURES

<u>Figure Number</u>	<u>Title</u>
1.1-1	Power/Flow Map
1.1-2	Deleted
2.1-1	APRM Flow Biased Scram and Rod Blocks
2.1-2	Deleted
4.1-1	Instrument Test Interval Determination Curves
4.2-2	Probability of System Unavailability Vs. Test Interval
3.4-1	Sodium Pentaborate Solution Volume Concentration Requirements
3.4-2	Saturation Temperature of Sodium Pentaborate Solution
3.6-1	DAEC Operating Limits
4.8.C-1	DAEC Emergency Service Water Flow Requirement
3.12-1	Flow-Dependent Minimum Critical Power Ratio ($MCPR_F$)
3.12-2	Power-Dependent Minimum Critical Power Ratio Multiplier (K_p)
3.12-3	Minimum Critical Power Ratio (MCPR) versus τ (Fuel Types: BP/P8X8R, GE8X8EB, LTA-311 and ELTA)
3.12-4	Limiting Average Planar Linear Heat Generation Rate (Fuel Type: BD303A)
3.12-5	Limiting Average Planar Linear Heat Generation Rate (Fuel Type: LTA 311)
3.12-6	Limiting Average Planar Linear Heat Generation Rate (Fuel Type BP/P8DRB301L)
3.12-7	Limiting Average Planar Linear Heat Generation Rate (Fuel Type: BD299A)
3.12-8	Limiting Average Planar Linear Heat Generation Rate (Fuel Types: BP/P8DRB299 and ELTA)
3.12-9	Limiting Average Planar Linear Heat Generation Rate (Fuel Type P8DRB284H)

<u>Figure Number</u>	<u>Title</u>
3.12-10	Flow-Dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Multiplier (MAPFAC _F)
3.12-11	Power-Dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Multiplier (MAPFAC _P)
3.12-12	Flow-Dependent Maximum Average Planar Linear Heat Generation Rate (MAPLHGR) Multiplier (MAPFAC _F) for SLO
6.2-1	DAEC Nuclear Plant Staffing

design pressure ($120\% \times 1150 = 1380$ psig; $120\% \times 1325 = 1590$ 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
<p>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.</p>	<p>b. (DELETED)</p>
<p>c. Control rods with inoperable accumulators or those whose position cannot be positively determined shall be considered inoperable.</p>	<p>c. Once per week when the plant is in operation, check status of pressure and level alarms for each CRD accumulator.</p>
<p>d. (DELETED)</p>	<p>d. Once per quarter verify that:</p> <ol style="list-style-type: none"> (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.
<p>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.</p>	<p>e. Once per cycle verify that:</p> <ol style="list-style-type: none"> (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.
<p>f. Inoperable control rods shall be positioned such that Specification 3.3.A.1 is met.</p>	

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
<p>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.</p> <p>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.</p> <p>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.</p> <p>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.</p>	<p>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.</p> <p>2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.</p> <p>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:</p> <p>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.</p> <p>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:</p> <p>1) The correctness of the Reduced Notch Worth Procedure sequence input to the RWM computer shall be verified.</p>

LIMITING CONDITION FOR OPERATION

SURVEILLANCE REQUIREMENT

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| | 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: | 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). |
| a. Both RBM channels shall be OPERABLE, or | |
| b. With one RBM channel inoperable, control rod withdrawal shall be blocked within 24 hours, unless OPERABILITY is restored within this time period, or | |
| c. With both RBM channels inoperable, control rod withdrawal shall be blocked until OPERABILITY of at least one channel is restored. | |

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 $\pm 10\%$ 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 OPERATION3.12 CORE THERMAL LIMITSApplicability

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 REQUIREMENTS4.12 CORE THERMAL LIMITSApplicability

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 $> 25\%$ 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 $< 25\%$ 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. Linear Heat Generation Rate (LHGR)

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.

B. Linear Heat Generation Rate (LHGR)

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 $< 25\%$ 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)

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 (τ). For core thermal power greater than or equal to 25% of rated and less than 30% of rated ($25\% < P < 30\%$), 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:
- a) The applicable flow-dependent MCPR ($MCPR_F$) determined from Figure 3.12-1, or
 - b) The appropriate RATED POWER MCPR from Figure 3.12-3 [$MCPR(100)$] multiplied by the applicable power-dependent MCPR multiplier

C. Minimum Critical Power Ratio (MCPR)

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.

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.
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 $< 25\%$ 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 a COLD SHUTDOWN condition within 36 hours.
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 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 become 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. Linear Heat Generation Rate (LHGR)

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 Δ CPR. The minimum Operating Limit MCPR of Specification 3.12.C bounds the sum of the Safety Limit MCPR and the largest Δ CPR.

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 (τ), 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: τ_{ave} = average scram insertion time to Notch 38, as measured by surveillance testing

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

τ_a = 67B scram insertion time to Notch 38

As the average scram time measured by surveillance testing (τ_{ave}), exceeds the ODYN Option B scram time (τ_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

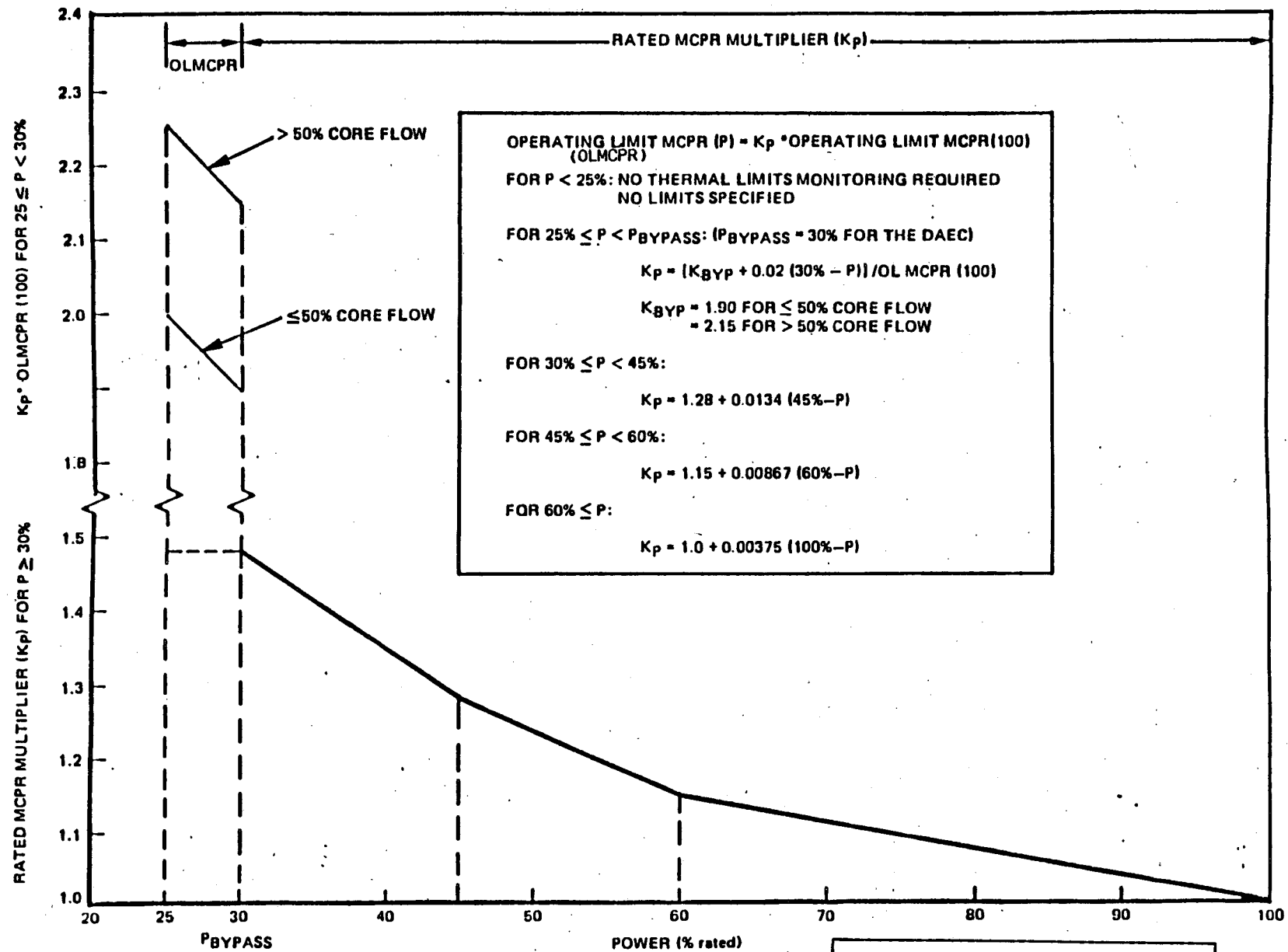
At core thermal power levels less than or equal to 25%, the reactor will be operating at minimum recirculation pump speed and the moderator void content will be very small. For all designated control rod patterns which may be employed at this point, operating plant experience indicated that the resulting 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.

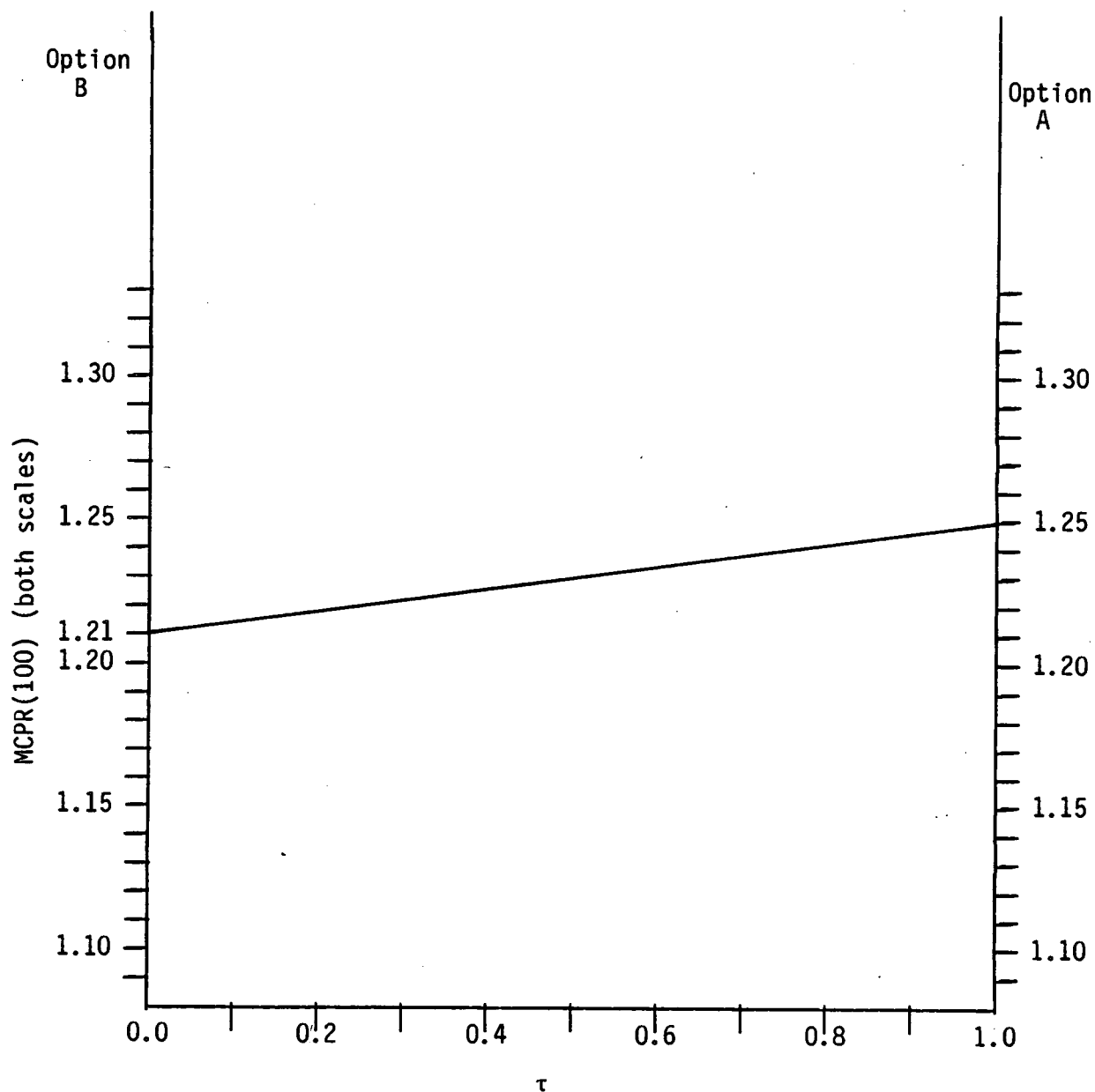
*Approved revision number at time reload fuel analyses are performed.

**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)

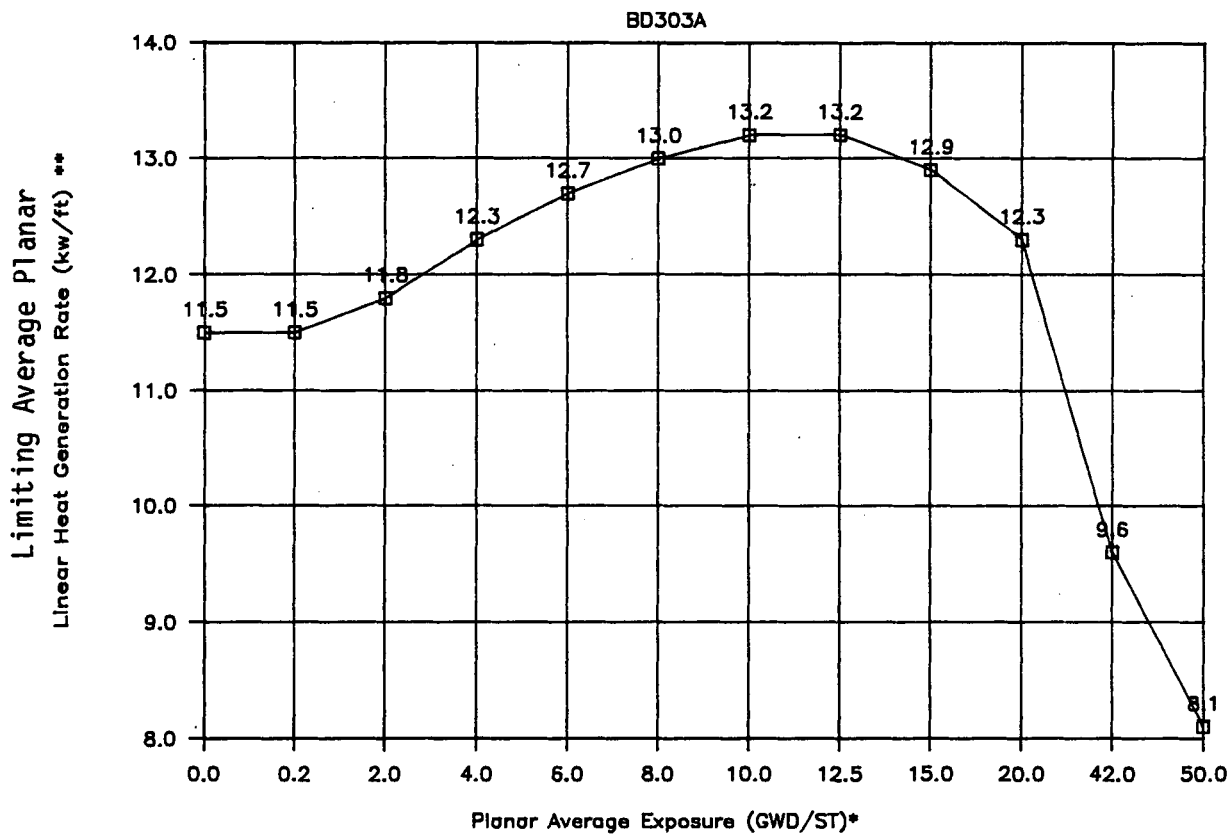
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TECHNICAL SPECIFICATIONS

MINIMUM CRITICAL POWER RATIO AT
RATED POWER [MCPR(100)] VERSUS τ

FUEL TYPES: BP/P8X8R, GE8X8EB,
LTA-311 and ELTA

FIGURE 3.12-3

MAPLHGR vs FUEL EXPOSURE



* 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.

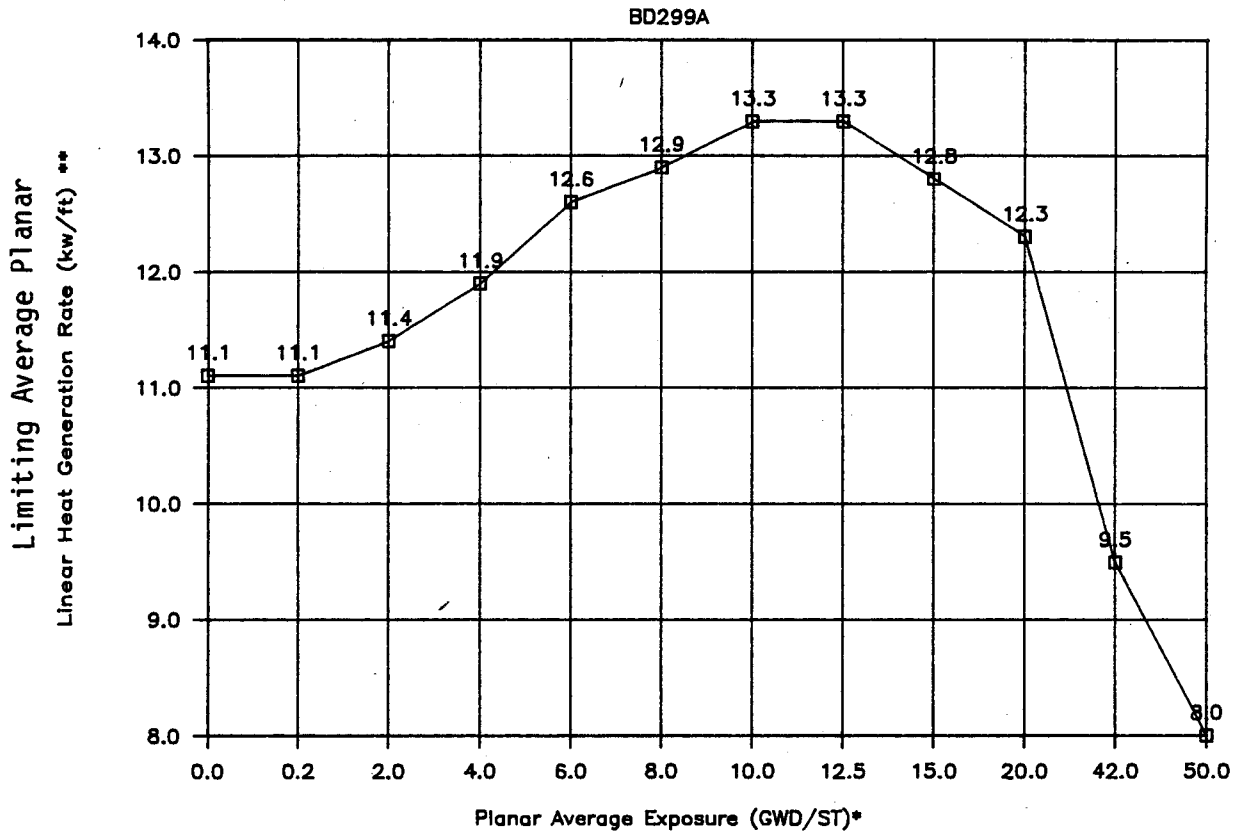
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TECHNICAL SPECIFICATIONS

LIMITING AVERAGE PLANAR LINEAR HEAT
GENERATION RATE AS A FUNCTION OF
PLANAR AVERAGE EXPOSURE

FUEL TYPE: BD303A

FIGURE 3.12-4

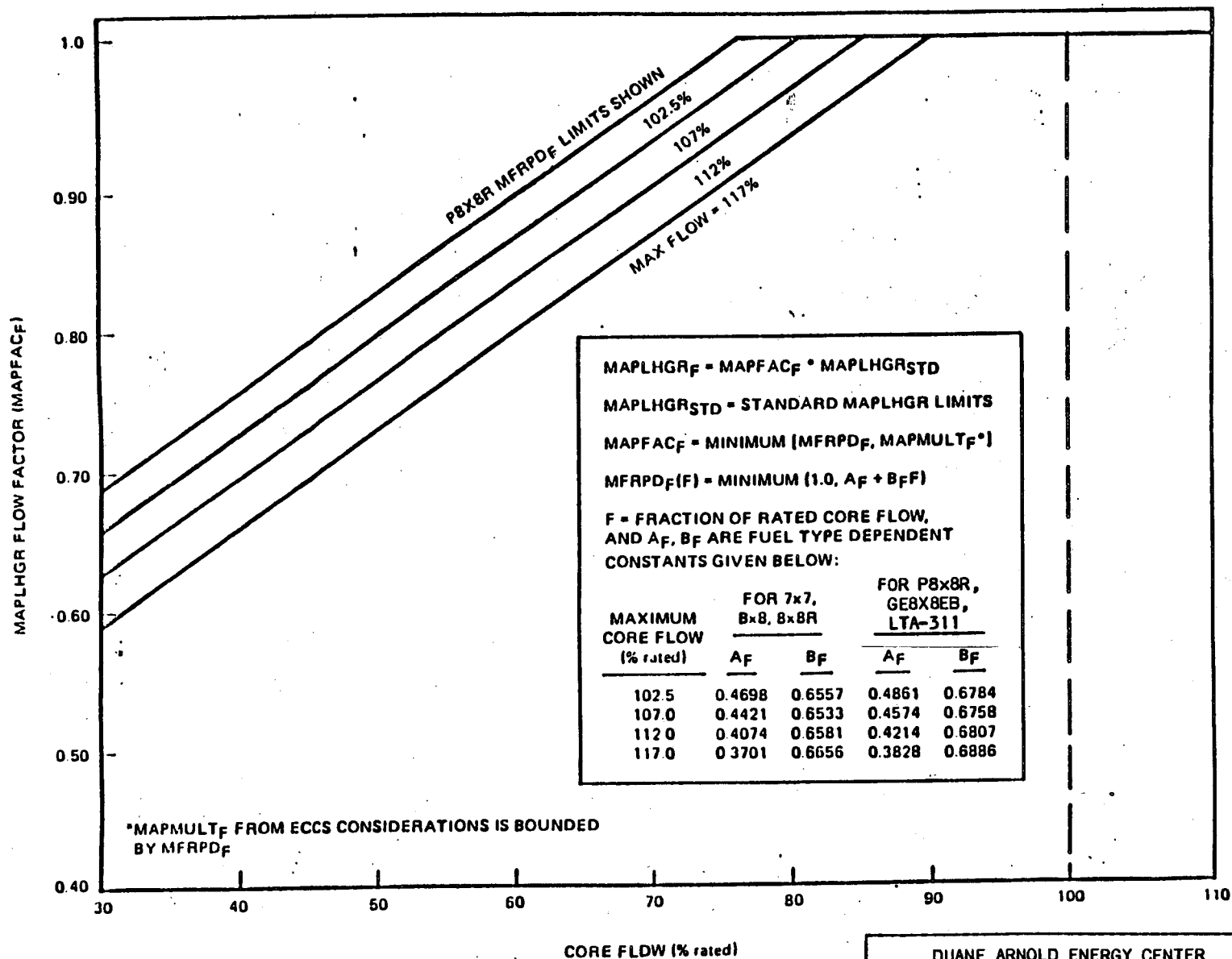
MAPLHGR vs FUEL EXPOSURE



* 1 Gwd/t = 1000 Mwd/t

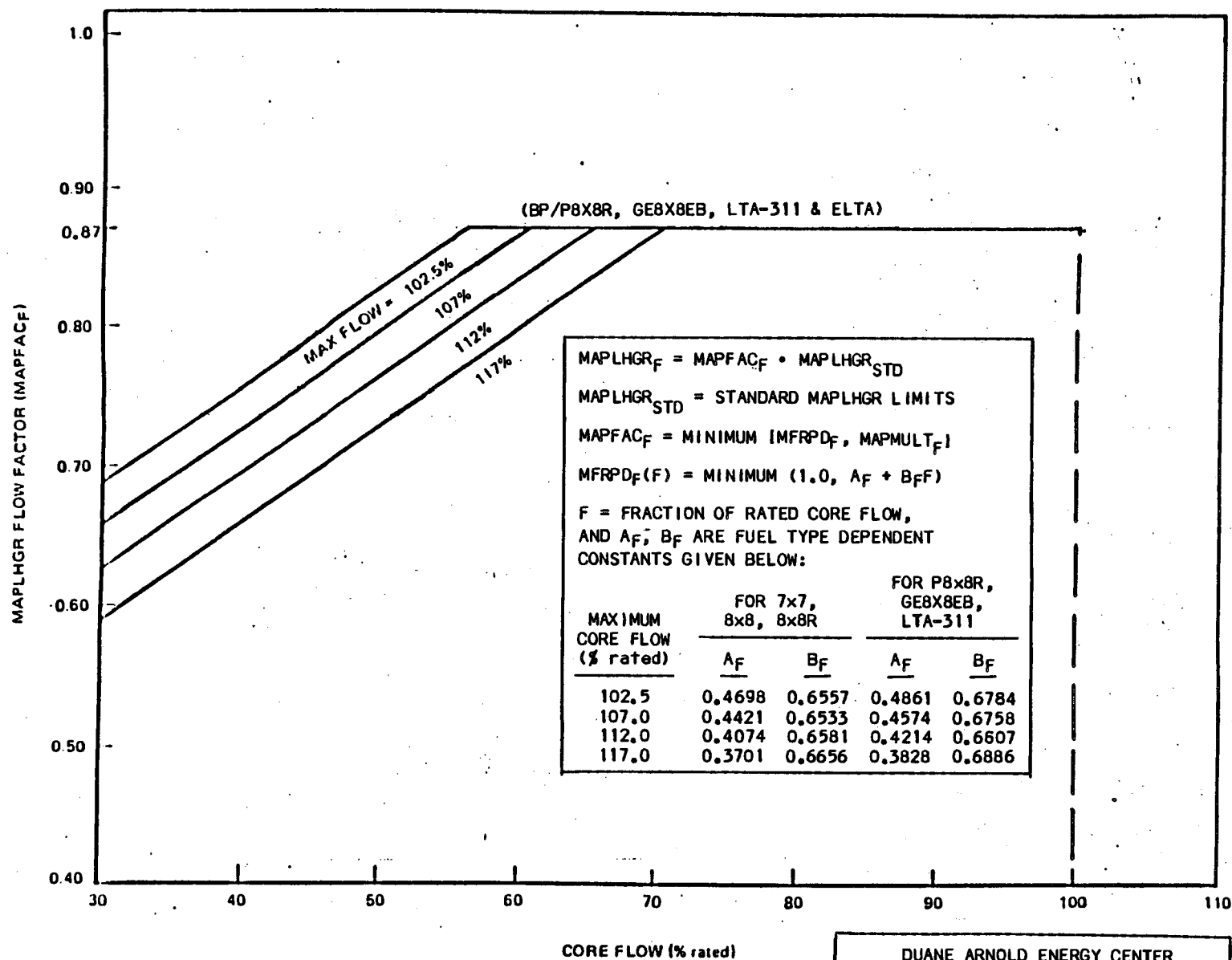
** These values are nominal values to be used for manual calculations. The actual lattice-type dependent values are modeled in the process computer.

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LIMITING AVERAGE PLANAR LINEAR HEAT GENERATION RATE AS A FUNCTION OF PLANAR AVERAGE EXPOSURE
FUEL TYPE: BD299A
FIGURE 3.12-7.



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



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FLOW-DEPENDENT MAXIMUM AVERAGE PLANAR
 LINEAR HEAT GENERATION RATE (MAPLHGR)
 MULTIPLIER (MAPFAC_F) FOR SLO
 FIGURE 3.12-12

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.