ENCLOSURE

Technical Specification Change Request No. 180
Supplement I

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C. Minimum Critical Power Ratio (MCPR)

During steady state power operation the MINIMUM CRITICAL POWER RATIO (MCPR) shall be equal to or greater than the MCPR limit as specified in the COLR.

When APRM status changes due to instrument failure (APRM or LPRM input failure), the MCPR requirement for the degraded condition shall be met within a time interval of eight (8) hours, provided that the control rod block is placed in operation during this interval.

For core flows other than rated, the nominal value for MCPR shall be increased by a factor of k_f, where k_f is as shown in the COLR.

If at any time during power operation it is determined by normal surveillance that the limiting value for MCPR is being exceeded for reasons other than instrument failure, action shall be initiated to restore operation to within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two [2] hours, action shall be initiated to bring the reactor to the cold shutdown condition within 36 hours. During this period, surveillance and corresponding action shall continue until reactor operation is within the prescribed limit at which time power operation may be continued.

Bases:

The Specification for average planar LHGR assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the 2200°F limit specified in 10 CFR 50.46. The analytical methods and assumptions used in evaluating the fuel design limits are presented in FSAR Chapter 4.

LOCA analyses are performed for each fuel design at selected exposure points to determined APLHGR limits that meet the PCT and maximum oxidation limits of 10 CFR 50.46. The analysis is performed using GE calculational models which are consistent with the requirements of 10 CFR 50, Appendix K. .

The PCT following a postulated LOCA is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is not strongly influenced by the rod to rod power distribution within an assembly. Since expected location variations in power distribution within a fuel assembly affect the calculated peak clad temperature by less than ± 20°F relative to the peak temperature for a typical fuel design, the limit on the average linear heat generation rate is sufficient to assure that calculated temperatures are below the limits specified in 10 CFR 50.46.

The maximum average planar LHGR limits for the various fuel types currently being used are provided in the COLR. The MAPLHGR limits for both five-loop and four-loop operation with the idle loop unisolated are shown. Four-loop operation with the idle loop isolated (suction, discharge and discharge bypass valves closed) requires that a MAPLHGR multiplier of 0.98 be applied to all fuel types. Additional requirements for isolated loop operation are given in Specification 3.3.F.2.

Fuel design evaluations are performed to demonstrate that the cladding 1% plastic strain and other fuel design limits are not exceeded during anticipated operational occurrences for operation with LHGRs up to the operating limit LHGR.

The analytical methods and assumptions used in evaluating the anticipated operational occurrences to establish the operating limit MCPR are presented in the FSAR, Chapters 4, 6 and 15 and in Technical Specification 6.9.1.f. To assure that the Safety Limit MCPR is not exceeded during any moderate frequency transient event, limiting transients have been analyzed to determine the largest reduction in Critical Power Ratio (CPR). The types of transients evaluated are pressurization, positive reactivity insertion and coolant temperature decrease. The operational MCPR limit is selected to provide margin to accommodate transients and uncertainties in monitoring the core operating state, manufacturing, and in the critical power correlation itself. This limit is derived by addition of the CPR for the most limiting transient to the safety limit MCPR designated in Specification 2.1.

The APRM response is used to predict when the rod block occurs in the analysis of the rod withdrawal error transient. The transient rod position at the rod block and corresponding MCPR can be determined. The MCPR has been evaluated for different APRM responses which would result from changes in the APRM status as a consequence of bypassed APRM channel and/or failed/bypassed LPRM inputs. The steady state MCPR required to protect the minimum transient CPR for the worst case APRM status condition (APRM Status 1) is determined in the rod withdrawal error transient analysis. The steady state MCPR values for APRM status conditions 1, 2, and 3 will be evaluated each cycle. For those cycles where the rod withdrawal error transient is not the most severe transient the MCPR Value for APRM status conditions 1, 2, and 3 will be the same and be equal to the limiting transient MCPR value.

(4) a summary of meteorological data collected during the year shall be included in the report submitted within 60 days after January 1 of each year. Alternatively, summary meteorological data may be retained by GPU Nuclear and made available to the NRC upon request.

e. Annual Radiological Environmental Report: A report of radiological environmental surveillance activities during each year shall be submitted before May 1 of the following year. Each report shall include the following information required in Specification 4.16 for radiological environmental surveillance:

(1) a summary description of the radiological environmental monitoring program,

(2) a map and a table of distances and directions (compass azimuth) of locations of sampling stations from the reactor,

(3) results of analyses of samples and of radiation measurements, (In the event some results are not available, the reasons shall be explained in the report. In the event the missing results are obtained, they shall be reported to the NRC as soon as is reasonable.)

(4) deviation(s) from the environmental sampling schedule in Table 4.16.1.

(5) identification of environmental samples analyzed when instrumentation was not capable of meeting detection capability in Table 4.16.2.

(6) a summary of the results of the land use survey.

(7) a summary of the results of licensee participation in an NRC approved inter-laboratory crosscheck program for environmental samples.

(8) results of dose evaluations to demonstrate compliance with 40 CFR Part 190.10a.

f. CORE OPERATING LIMITS REPORT (COLR)

1. Core operating limits shall be established prior to each reload cycle, or prior to any remaining portion of a reload cycle for the following:

a. The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) for Specification 3.10.A

b. The K_f core flow adjustment factor for Specification 3.10.C.

c. The MINIMUM CRITICAL POWER RATIO (MCPR) for Specification 3.10.C

d. The LOCAL LINEAR HEAT GENERATION RATE (LLHGR) for Specification 3.10.B.

and shall be documented in the COLR.

2. The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by the NRC, specifically those described in the following documents.

   a. GPU Nuclear (GPUN) Topical Report (TR) 020, Methods for the Analysis of Boiling Water Reactors Lattice Physics, latest approved revision

   b. GPUN TR 021, Methods for the Analysis of Boiling Water Reactors Steady State Physics, latest approved revision

   c. GPUN TR 033, Methods for the Generation of Core Kinetics Data for RETRAN-02, latest approved revision

   d. GPUN TR 040, Steady-State and Quasi-Steady-State Methods Used in the Analysis of Accidents and Transients, latest approved revision

   e. GPUN TR 045, BWR-2 Transient Analysis Model Using the Retran Code, latest approved revision

   f. NEDE 31462P and NEDE-31462, Oyster Creek Nuclear Generating Station SAFER/CORECOOL/GESTR-LOCA Loss-of-Coolant Accident Analysis, latest approved revision

   g. NEDE-24011, General Electric Standard Application for Reactor Fuel, latest approved revision

   h. NEDE-24195, General Electric Reload Fuel Application for Oyster Creek, latest approved revision

   i. XN-75-55-(A); XN-75-55, Supplement 1-(A); XN-75-55, Supplement 2-(A), Revision 2, "Exxon Nuclear Company WREM-Based NJP-BWR ECCS Evaluation Model and Application to the Oyster Creek Plant," April 1977

   j. XN-75-36(NP)-(A); XN-75-36(NP), Supplement 1-(A), "Spray Cooling Heat Transfer Phase Test Results, ENC - 8x8 BWR Fuel 60 and 63 Active Rods, Interim Report, " October 1975

3. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as shutdown margin, transient analysis limits, and accident analysis limits) of the safety analysis are met.

4. The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements shall be provided, upon issuance for each reload cycle, to the NRC Document Control Desk with copies to the Regional Administrator and Resident Inspector.

Basis: 6.9.1.e

An annual report of radiological environmental surveillance activities includes factual data summarizing results of activities required by the surveillance program. In order to aid interpretation of the data, GPUN may choose to submit analysis of trends and comparative non regional radiological environmental data. In addition, the licensee may choose to discuss previous radiological environmental data as well as the observed radiological environmental impacts of station operation (if any) on the environment.

6.9.2 REPORTABLE EVENTS

The submittal of Licensee Event Reports shall be accomplished in accordance with the requirements set forth in 10 CFR 50.73.