2.0 SAFETY LIMITS

2.1 Safety Limits

2.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% of rated core flow:

THERMAL POWER shall be ≤ 25% of RATED THERMAL POWER.

2.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% of rated core flow:

MINIMUM CRITICAL POWER RATIO shall be ≥ 1.06.

2.1.3 Whenever the reactor is in the cold shutdown condition with irradiated fuel in the reactor vessel, the water level shall not be less than 12 inches above the top of the normal active fuel zone.

2.1.4 Reactor steam dome pressure shall be ≤ 1325 psig at any time when irradiated fuel is present in the reactor vessel.

2.2 Safety Limit Violation

With any Safety Limit not met the following actions shall be met:

2.2.1 Within one hour notify the NRC Operations Center in accordance with 10CFR50.72.

2.2.2 Within two hours:

A. Restore compliance with all Safety Limits, and

B. Insert all insertable control rods.

2.2.3 The Station Director and Senior Vice President - Nuclear and the Nuclear Safety Review and Audit Committee (NSRAC) shall be notified within 24 hours.

2.2.4 A Licensee Event Report shall be prepared pursuant to 10CFR50.73. The Licensee Event Report shall be submitted to the Commission, the Operations Review Committee (ORC), the NSRAC and the Station Director and Senior Vice President - Nuclear within 30 days of the violation.

2.2.5 Critical operation of the unit shall not be resumed until authorized by the Commission.

FUEL CLADDING INTEGRITY (2.1.1) (Cont)	Since the pressure drop in the bypass region is essentially all elevation head, the core pressure drop at low power and flows will always be greater than 4.5 psi. Analyses show that with a bundle flow of 28 x 10^3 lbs/hr, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.5 psi driving head will be greater than 28 x 10^3 lbs/hr. Full scale ATLAS test data taken at pressures from 14.7 psia to 800 psia indicate that the fuel assembly critical power at this flow is approximately 3.35 MWt. With the design peaking factors, this corresponds to a THERMAL POWER of more than 50% of RATED THERMAL POWER. Thus, a THERMAL POWER limit of 25% of RATED THERMAL POWER for reactor pressure below 785 psig is conservative.
MINIMUM CRITICAL POWER RATIO (2.1.2)	The Safety Limit MCPR is determined using the General Electric Thermal Analysis Basis, GETAB (2), which is a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. Instead of the standard GETAB model uncertainties, revised uncertainties in accordance with references 3 and 4 were used to calculate the SLMCPR. The probability of the occurrence of boiling transition is determined using the General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation. The range of validity of the GEXL correlation is specified in References 5 and 6. The fuel cladding integrity Safety Limit is set such that no fuel damage is calculated to occur if the limit is not violated. Since the parameters which result in fuel damage are not directly observable during reactor operation, the thermal and hydraulic conditions resulting in a departure from nucleate boiling have been used to mark the beginning of the region where fuel damage could occur. Although it is recognized that a departure from nucleate boiling would not result in damage to BWR fuel rods, the critical power at

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MINIMUM CRITICAL POWER RATIO (2.1.2) (Cont)	which boiling transition is calculated to occur has been adopted as a convenient limit. However, the uncertainties in monitoring the core operating state and in the procedures used to calculate the critical power result in an uncertainty in the value of the critical power. Therefore, the fuel cladding integrity Safety Limit is defined as the CPR in the limiting fuel assembly for which more than 99.9% of the fuel rods in the core are expected to avoid boiling transition considering the power distribution within the core and all uncertainties. The Safety Limit MCPR is determined using a statistical model that combines all of the uncertainties in operating parameters and the procedures used to calculate critical power. The probability of the occurrence of boiling transition is determined using the approved General Electric Critical Power correlations. Details of the fuel cladding integrity Safety Limit calculation are given in Reference 1. References 3 and 4 include a tabulation of the uncertainties used in the determination of the Safety Limit MCPR and of the nominal values of the parameters used in the Safety Limit MCPR statistical analysis.
REACTOR WATER LEVEL (Shutdown Condition) (2.1.3)	With fuel in the reactor vessel during periods when the reactor is shutdown, consideration must be given to water level requirements due to the effect of decay heat. If reactor water level should drop below the top of the active fuel during this time, the ability to cool the core is reduced. This reduction in core cooling capability could lead to elevated cladding temperatures and clad perforation. The core can be cooled sufficiently should the water level be reduced to two-thirds the core height. Establishment of the safety limit at 12 inches above the top of the fuel provides adequate margin. This level will be continuously monitored.

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REACTOR STEAM DOME PRESSURE (2.1.4)

The Safety Limit for the reactor steam dome pressure has been selected such that it is at a pressure below which it can be shown that the integrity of the system is not endangered. The reactor pressure vessel is designed to Section III of the ASME Boiler and Pressure Vessel Code (1965 Edition, including the January 1966 Addendum), which permits a maximum pressure transient of 110%, 1375 psig, of design pressure 1250 psig. The Safety Limit of 1325 psig, as measured by the reactor steam dome pressure indicator, is equivalent to 1375 psig at the lowest elevation of the reactor coolant system. The reactor coolant system is designed to the USAS Nuclear Power Piping Code, Section B31.1.0 for the reactor recirculation piping, which permits a maximum pressure transient of 120% of design pressures of 1148 psig at 562°F for suction piping and 1241 psig at 562°F for discharge piping. The pressure Safety Limit is selected to be the lowest transient overpressure allowed by the applicable codes.

REFERENCES

1) "General Electric Standard Application for Reactor Fuel," NEDE-24011-P-A (through the latest approved amendment at the time the reload analyses are performed as specified in the CORE OPERATING LIMITS REPORT).

2) General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, January 1977, NEDE-10958-PA and NEDO-10958-A.

3) "Methodology & Uncertainties for SLMCPR Evaluations," NEDC-32601-P-A (August 1999).

4) "Power Distribution Uncertainties for Safety Limit MCPR Evaluations," NEDC-32694-P-A (August 1999).

5) "GE 11 Compliance with Amendment 22 of GESTAR II," NEDE-31917P (April 1991).

6) "GE 14 Compliance with Amendment 22 of GESTAR II," NEDC-32868P (December 1998).

3.11 REACTOR FUEL ASSEMBLY

A. Average Planar Linear Heat Generation Rate (APLGHR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in 10CFR50, Appendix K.

The analytical method used to determine the APLHGR limiting values is described in the topical reports listed in Specification 5.6.5.b.

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.

The analytical method used to determine the LHGR limiting value is described in the topical reports listed in Specifications 5.6.5.b.

C. Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

For any abnormal operating transient analysis with the initial condition of the reactor 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 the instrument trip settings given in Tables 3.1.1, 3.2.A and 3.2.B.

The analytical method used to determine the Operating Limit MCPR values in the CORE OPERATING LIMITS REPORT is described in the topical reports listed in Specification 5.6.5.b. By maintaining MCPR greater than or equal to the Operating Limit MCPR, the Safety Limit MCPR specified in Specification 2.1.2 is maintained in the event of the most limiting abnormal operating transient.

D. Power/Flow Relationship During Power Operation

The power/flow curve is the locus of core thermal power as a function of flow from which the occurrence of abnormal operating transients will yield results within defined plant safety limits. Each transient and postulated accident applicable to operation of the plant was analyzed along the power/flow line. The analysis justifies the operating envelope bounded by the power/flow curve as long as other operating limits are satisfied. Operation under the power/flow line is designed to enable the direct ascension to full power within the design basis for the plant.

5.6 Reporting Requirements

5.6.3 Radioactive Effluent Release Report

The Radioactive Effluent Release Report covering the operation of the unit shall be submitted in accordance with 10 CFR 50.36a by May 15th of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be consistent with the objectives outlined in the ODCM and process control procedures and in conformance with 10 CFR 50.36a and 10 CFR 50, Appendix I, Section IV.B.1.

5.6.4 Monthly Operating Reports

Routine reports of operating statistics and shutdown experience shall be submitted on a monthly basis no later than the 15th of each month following the calendar month covered by the report.

5.6.5 Core Operating Limits Report (COLR)

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

1. Table 3.1.1 – APRM High Flux trip level setting
2. Table 3.2.C –APRM Upscale trip level setting
3. 3.11.A – Average Planar Linear Heat Generation Rate (APLHGR)
4. 3.11.B – Linear Heat Generation Rate (LHGR)
5. 3.11.C –Minimum Critical Power Ratio (MCPR)
6. 3.11.D – Power/Flow Relationship During Power Operation

b. 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:

1. NEDE-24011-P-A, "General Electric Standard Application for Reactor Fuel," (through the latest approved amendment at the time the reload analyses are performed as specified in the COLR).

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5.6.5

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

d. The COLR, including any midcycle revisions or supplements, shall be provided upon issuance for each reload cycle to the NRC.