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Karl R. Goller, Assistant Director for Operating Reactors, RL

MONTICELLO TECHNICAL SPECIFICATION BASES FOR THE THERMAL LIMITS

Plant Name:	Monticello
Docket No.:	50-263
Responsible Branch and Project Leader:	ORB-2 B. Buckley
Technical Review Branch Involved:	Reactor Systems Branch
Review Status:	Complete

Enclosed are the Technical Specification bases for the Monticello Nuclear Generating Plant which set forth the bases for the established thermal limits of Specifications 2.1, 3.11C, and 4.11C. Also, the bases for Specification 3.11A, entitled "Average Planar Linear Heat Generation Rate" (APLHGR) are provided.

Note that the Technical Specification bases for the thermal limits were applied to the Monticello Cycle 4 safety analysis. As stated by the licensee, the Cycle 5 analysis was found to be bounded by the Cycle 4 analysis. Therefore, the enclosed bases justify the Technical Specification thermal limits for Cycle 5 plant operation.

Victor Stello, Jr., Assistant Director
for Reactor Safety
Division of Technical Review
Office of Nuclear Reactor Regulation

Enclosure:
Tech. Spec. Bases

cc: S. Hanauer
R. Heineman
R. Boyd
V. Stello
D. Ziemann
B. Buckley
P. Check
T. Novak
W. Minners
R. Woods
A. Ignatonis

ENCLOSURE TO MONTICELLO

TECHNICAL SPECIFICATION BASES

Bases:

2.1 Fuel Cladding Integrity

A. Fuel Cladding Integrity Limit at Reactor Pressure >800 psia and Core Flow >10% of Rated

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 necessarily result in damage to BWR fuel rods, the critical power at 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 procedure 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 critical power ratio 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 the General Electric Thermal Analysis Basis, GETAB[1], which is 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 General Electric Critical Quality (X) - Boiling Length (L), GEXL, correlation.

The GEXL correlation is valid over the range of conditions used in the tests of the data used to develop the correlation. These conditions are:

Pressure:	800 to 1400 psia	
Mass flux:	0.1 to 1.25 10^6 lb/hr-ft^2	
Inlet Subcooling:	0 to 100 Btu/lb	
Local Peaking:	1.61 at a corner rod to	
	1.47 at an interior rod	
Axial Peaking:	Shape	Max/Avg.
	Uniform	1.0
	Outlet Peaked	1.60
	Inlet Peaked	1.60
	Double Peak	1.46 and 1.38
	Cosine	1.39
Rod Array	16, 64 Rods in an 8 x 8 array	
	49 Rods in a 7 x 7 array	

The required input to the statistical model are the uncertainties listed on Table 2.1-1, the nominal values of the core parameters listed in Table 2.1-2, and the relative assembly power distribution shown in Table 2.1-3. Table 2.1-4 shows the R-factor distributions that are input to the statistical model which is used to establish the safety limit MCPR. The R-factor distributions shown are taken near the beginning of the fuel cycle.

The bases or the uncertainties in the co. parameters are given in NEDO-20340,[2] and the basis for the uncertainty in the GEXL correlation is given in NEDO-10958[1]. The power distribution is based on a typical 764 assembly core in which the rod pattern was arbitrarily chosen to produce a skewed power distribution having the greatest number of assemblies at the highest power levels. The worst distribution in Millstone Unit 1 during any fuel cycle would not be as severe as the distribution used in the analysis.

B. Core Thermal Power Limit (Reactor Pressure ≤ 800 psia on Core Flow ≤ 10% of Rated)

The use of the GEXL correlation is not valid for the critical power calculations at pressures below 800 psia or core flows less than 10% of rated. Therefore, the fuel cladding integrity safety limit is established by other means. This is done by establishing a limiting condition of core thermal power operation with the following basis.

Since the pressure drop in the bypass region is essentially all elevation head which is 4.56 psi the core pressure drop at low power and all flows will always be greater than 4.56 psi. Analyses show that with a flow of 28×10^3 lbs/hr bundle flow, bundle pressure drop is nearly independent of bundle power and has a value of 3.5 psi. Thus, the bundle flow with a 4.56 psi driving head will be greater than 28×10^3 lbs/hr irrespective of total core flow and independent of bundle power for the range of bundle powers of concern. 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 the 3.35 MWt bundle power corresponds to a core thermal power of more than 50%. Therefore a core thermal power limit of 25% for reactor pressures below 800 psia or core flow less than 10% is conservative.

C. Power Transient

Plant safety analyses have shown that the scrams caused by exceeding any safety setting will assure that the Safety Limit of Specification 2.1.A or 2.1.B will not be exceeded. Scram times are checked periodically to assure the insertion times are adequate. The thermal power transient resulting when a scram is accomplished other than by the expected scram signal (e.g., scram from neutron flux following closure of the main turbine stop valves) does not necessarily cause fuel damage. However, for this specification a Safety Limit violation will be assumed when a scram is only accomplished by means of a backup feature of the plant design. Th concept of not approaching a Safety Limit provided scram signals are operable is supported by the extensive plant safety analysis.

The computer provided with Monticello has a sequence annunciation program which will indicate the sequence in which events such as scram, APRM trip initiation, pressure scram initiation, etc. occur. This program also indicates when the scram setpoint is cleared. This will provide information on how long a scram condition exists and thus provide some measure of the energy added during a transient.

D. Reactor Water Level (Shutdown Condition)

During periods when the reactor is shutdown, consideration must also 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.

References

1. General Electric Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, General Electric Co. BWR Systems Department, November 1973 (NEDO-10958).

2. Process Computer Performance Evaluation Accuracy, General Electric Company BWR Systems Department, June, 1974 (NEDO-20340).

Table 2.1-1

UNCERTAINTIES USED IN THE DETERMINATION

OF THE FUEL CLADDING SAFETY LIMIT

Quantity	Standard Deviation (% of Point)
Feedwater Flow	1.76
Feedwater Temperature	0.76
Reactor Pressure	0.5
Core Inlet Temperature	0.2
Core Total Flow	2.5
Channel Flow Area	3.0
Friction Factor Multiplier	10.0
Channel Friction Factor Multiplier	5.0
TIP Readings	8.7
R Factor	1.6
Critical Power	3.6

Table 2.1-2

NOMINAL VALUES OF PARAMETERS USED IN

THE STATISTICAL ANALYSIS OF FUEL CLADDING INTEGRITY SAFETY LIMIT

Core Thermal Power	3293 MW
Core Flow	102.5 Mlb/hr
Dome Pressure	1010.4 psig
Channel Flow Area	0.1078 ft^2
R-Factor	1.098 (7x7)
	1.100 (8x8)

Table 2.1-3

RELATIVE BUNDLE POWER DISTRIBUTION

USED IN THE GETAB STATISTICAL ANALYSIS

Range of Relative Bundle Power	Percent of Fuel Bundles Within Power Interval
1.375 to 1.425	6.6
1.325 to 1.375	3.2
1.275 to 1.325	15.6
1.225 to 1.275	10.8
1.175 to 1.225	6.6
1.125 to 1.175	4.9
1.075 to 1.125	9.0
1.025 to 1.075	4.0
0.175 to 1.025	39.3
	Sum = 100

Table 2.1-4

R-FACTOR DISTRIBUTION USED IN GETAB STATISTICAL ANALYSIS

7x7 Rod Array		8x8 Rod Array	
R-Factor	Rod Sequence No.	R-Factor	Rod Sequence No.
1.098	1	1.100	1
1.083	2	1.100	2
1.075	3	1.095	3
1.062	4	1.095	4
1.052	5	1.093	5
1.042	6	1.093	6
1.042	7	1.092	7
≤ 1.027	8 thru 49	≤ 1.077	8 thru 63

BASES

3.11A Average Planar Linear Heat Generation Rate (APLHGR)

This specification assures that the peak cladding temperature following the postulated design basis loss-of-coolant accident will not exceed the limit specified in the 10 CFR 50, Appendix K.

The peak cladding temperature (PCT) following a postulated loss-of-coolant accident is primarily a function of the average heat generation rate of all the rods of a fuel assembly at any axial location and is only dependent, secondarily on the rod to rod power distribution within an assembly. The peak clad temperature is calculated assuming a LHGR for the highest powered rod which is equal to or less than the design LHGR corrected for densification. This LHGR times 1.02 is used in the heat-up code along with the exposure dependent steady state gap conductance and rod-to-rod local peaking factors. The Technical Specification APLHGR is this LHGR of the highest powered rod divided by its local peaking factor. The limiting value for APLHGR is shown in Figure 3.11.1.

The calculational procedure used to establish the APLHGR shown on Figure 3.11.1 is based on a loss-of-coolant accident analysis. The analysis was performed using General Electric (GE) calculational models which are consistent with the requirements of Appendix K to 10 CFR 50. A complete discussion of each code employed in the analysis is presented in Reference 1. Differences in this analysis as compared to previous analyses performed with Reference 1 are: (1) The analyses assumes a fuel assembly planar power consistent with 102% of the MAPLHGR shown in Figure 3.11.1, (2) Fission product decay is computed assuming an energy release rate of 200 MEV/Fission; (3) Pool boiling is assumed after nucleate boiling is lost during the flow stagnation period; (4) The effects of core spray entrainment and counter-current flow limiting as described in Reference 2, are included in the reflooding calculations.

A list of the significant plant input parameters to the loss-of-coolant accident analysis is presented in Table 1.

TABLE 1

SIGNIFICANT INPUTS PARAMETERS TO THE LOSS-OF-COOLANT ACCIDENT ANALYSIS FOR MONTICELLO

PLANT PARAMETERS:

Core Thermal Power.................... 1703 MWt which corresponds to 102% of licensed core power*

Vessel Steam Output............ 6.913×10^6 Lbm/h which corresponds to 102 % of rated steam flow

Vessel Steam Dome Pressure............ 1040 psia

Design Basis Recirculation Line Break Area for Large Breaks 3.9 ft^2 1.0 ft^2

Recirculation Line Break Area for Small Breaks 1.0 ft^2 0.07 ft^2

FUEL PARAMETERS:

FUEL TYPE	FUEL BUNDLE GEOMETRY	PEAK TECHNICAL SPECIFICATION LINEAR HEAT GENERATION RATE (kw/ft)	DESIGN AXIAL PEAKING FACTOR	INITIAL MINIMUM CRITICAL POWER RATIO
Initial Core 7D225	7 x 7	17.5	1.57	1.18
Reload 1 7D230	7 x 7	17.5	1.57	1.18
Reload 2 8D262	8 x 8	13.4	1.57	1.18
Reload 3 8D250	8 x 8	13.4	1.57	1.18
Reload 4 8D219	8 x 8	13.4	1.57	1.18

A more detailed list of input to each model and its source is presented in Section II of Reference 1.

*This power level equals the Appendix K requirement of 102%. The core heatup calculation assumes a bundle power consistent with operation of the highest powered rod at 102% of its maximum (technical specification) linear heat generation rate.

REFERENCES

1. General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K, NEDE-20566 (Draft), submitted August 1974.

2. General Electric Refill Reflood Calculation (Supplement to SAFE Code Description) transmitted to USAEC by letter, G. L. Gyorey to V. Stello, Jr., dated December 20, 1974.

Bases:

3.11C Minimum Critical Power Ratio (MCPR)

Operating Limit MCPR

The required operating limit MCPR's at steady state operating conditions as specified in Specification 3.11C are derived from the established fuel cladding integrity Safety Limit MCPR of 1.06, and an analysis of abnormal operational transients[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 setting given in Specification 2.3.

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 turbine trip with failure of the turbine bypass. This transient yields the largest Δ MCPR. When added to the Safety Limit MCPR of 1.06 the required minimum operating limit MCPR of specification 3.11C is obtained.

Prior to the analysis of abnormal operational transients an initial fuel bundle MCPR was determined. This parameter is based on the bundle flow calculated by a GE multi-channel steady state flow distribution model as described in Section 4.4 of NEDO-20360[3] and on core parameters shown in Reference 2 (response to Items 2 and 9).

The evaluation of a given transient begins with the system initial parameters shown in Table 6-1 (page 6-10) of Reference 1 that are input to a GE core dynamic behavior transient computer program described in NEDO-10802[3]. Also, the void reactivity coefficients that were input to the transient calculational procedure are based on a new method of calculation termed NEV which provides a better agreement between the

calculated and plant instrument power distributions. The outputs of this program along with the initial MCPR form the input for further analyses of the thermally limiting bundle with the single channel transient thermal hydraulic SCAT code described in NEDE-20566.(4) The principal result of this evaluation is the reduction in MCPR caused by the transient.

MCPR Limits for Core Flows Other than Rated

The purpose of the K_f factor is to define operating limits at other than rated flow conditions. At less than 100% flow the required MCPR is the product of the operating limit MCPR and the K_f factor. Specifically, the K_f factor provides the required thermal margin to protect against a flow increase transient. The most limiting transient initiated from less than rated flow conditions is the recirculation pump speed up caused by a motor-generator speed control failure.

For operation in the automatic flow control mode, the K_f factors assure that the operating limit MCPR of Specification 3.11C will not be violated should the most limiting transient occur at less than rated flow. In the manual flow control mode, the K_f factors assure that the Safety Limit MCPR will not be violated for the same postulated transient event.

The K_f factor curves shown in Figure 3.11.2 were developed generically which are applicable to all BWR/2, BWR/3, and BWR/4 reactors. The K_f factors were derived using the flow control line corresponding to rated thermal power at rated core flow.

For the manual flow control mode, the K_f factors were calculated such that at the maximum flow state (as limited by the pump scoop tube set point) and the corresponding core power (along the rated flow control line), the limiting bundle's relative power was

adjusted until the MCPR was slightly above the Safety Limit. Using this relative bundle power, the MCPR's were calculated at different points along the rated flow control line corresponding to different core flows. The ratio of the MCPR calculated at a given point of core flow, divided by the operating limit MCPR determines the K_f.

For operation in the automatic flow control mode, the same procedure was employed except the initial power distribution was established such that the MCPR was equal to the operating limit MCPR at rated power and flow.

The K_f factors shown in Figure 3.11.2, are conservative for the Monticello Nuclear Generating Plant operation because the operating limit MCPR's of Specification 3.11C are greater than the original 1.20 operating limit MCPR used for the generic derivation of K_f.

4.11C 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 mode relative to MCPR. During initial start-up testing of the plant, a MCPR evaluation will be made at 25% thermal power level with minimum recirculation pump speed. The MCPR margin will thus be demonstrated such that future MCPR evaluation below this power level will be shown to be unnecessary. 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 a limiting control rod pattern is approached ensures that MCPR will be known following a change in power or power shape (regardless of magnitude) that could place operation at a thermal limit.