Exhibit B

Monticello Nuclear Generating Plant

<u>License Amendment Request dated July 27, 1987</u>

Evaluation of Proposed Changes

Exhibit B consists of marked up pages for the Monticello Nuclear Generating Plant Technical Specifications showing the proposed changes as listed below:

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114b (two pages)
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3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
Any four rod group may contain a control rod which is valved out of service provided the above requirements and Specification 3.3.A are met.	
Delete → 3. If the cycle average scram insertion time (τ_{AVG}), based on the de-energization of the scram pilot valve solenoids at time zero, of all operable control rods in the reactor power operation condition at the 20% inserted position is larger than the adjusted analysis mean scram time (τ_{B}), a more restrictive MCPR limit (see section 3.11.C) shall be used.	
D. Control Rod Accumulators	D. Control Rod Accumulators
In the "Startup" or "Run" Mode, a rod accumulator may be inoperable provided that no other control rod in the nine-rod square array around this rod has a:	Once a shift check the status in the control room of the required Operable accumulator pressure and level alarms.
1. Inoperable accumulator.	
2. Directional control valve electrically disarmed while in a non-fully inserted position.	
If a control rod with an inoperable accumulator is inserted "full-in" and its directional control valves are electrically disarmed, it shall not be considered to have an inoperable accumulator.	
In the "Refuel" Mode, the accumulator associated with any withdrawn control rod must be Operable unless all the fuel has been removed from the cell containing that control rod.	

Bases Continued 3.3 and 4.3:

The analysis assumes 50 milliseconds for Reactor Protection System delay, 200 milli seconds from de-energization of scram solenoids to the beginning of rod motion, and 175 milliseconds later the rods are at the 5% position.

Section 3.3.C.3 allows a lower MCPR limit to be used if the cycle average scram time (τ_{AVE}) is less than the adjusted analysis mean scram time (τ_B) (see Reference 7, of Section 3.11)

τ_{AVE} is the weighted cycle average scram time to the 20% insertion position (∿ notch 38) of all the operable rods measured at any point in the cycle.

$$\tau_{AVE} = \frac{\sum_{i=1}^{n} N_i \tau_i}{\sum_{i=1}^{n} N_i}$$

where: n = the number of surveillance tests performed to date in this cycle.

N_i = number of control rods measured in the ith test.

τ_i = average scram time to the 20% insertion position of all rods measured in the i th test.

τ_B is the adjusted analysis mean scram time to the 20% insertion position.

$$\tau_B = 0.710 + 0.0875 \left(\frac{N_1}{\sum_{i=1}^{n} N_i} \right)^{\frac{1}{2}}$$

where: N_i = total number of active rods measured in the first test following core alterations.

0.710 = the mean scram time used in the analysis.

0.0875 = 1.65x0.053 where 1.65 is the appropriate statistical number to provide a 95% confidence level and, 0.053 is the standard deviation of the distribution 20% position, that was used in the analysis.

Delete

delete and replace with attached

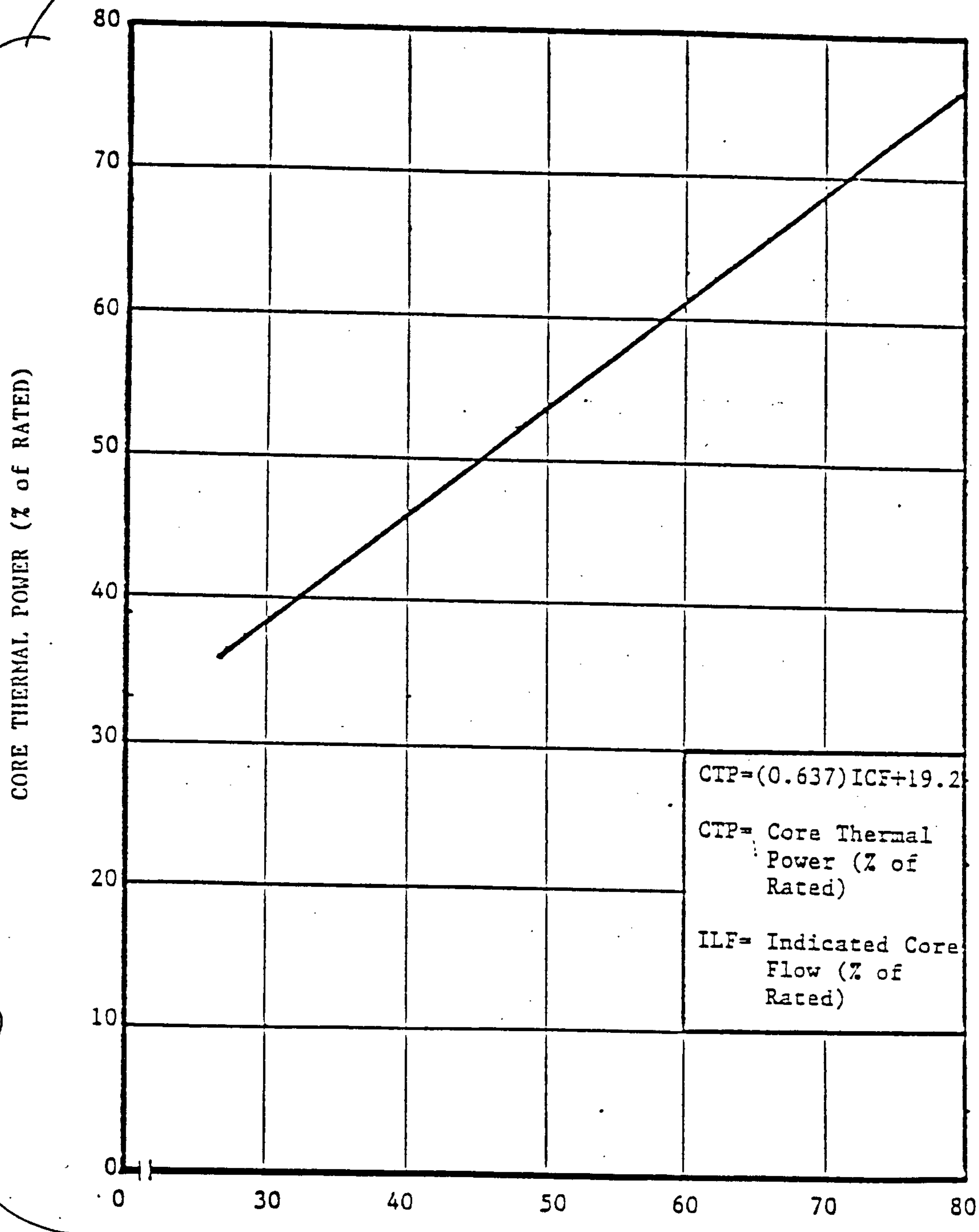

Figure 3.5.1 Single Loop Operation Surveillance Power/Flow Curve

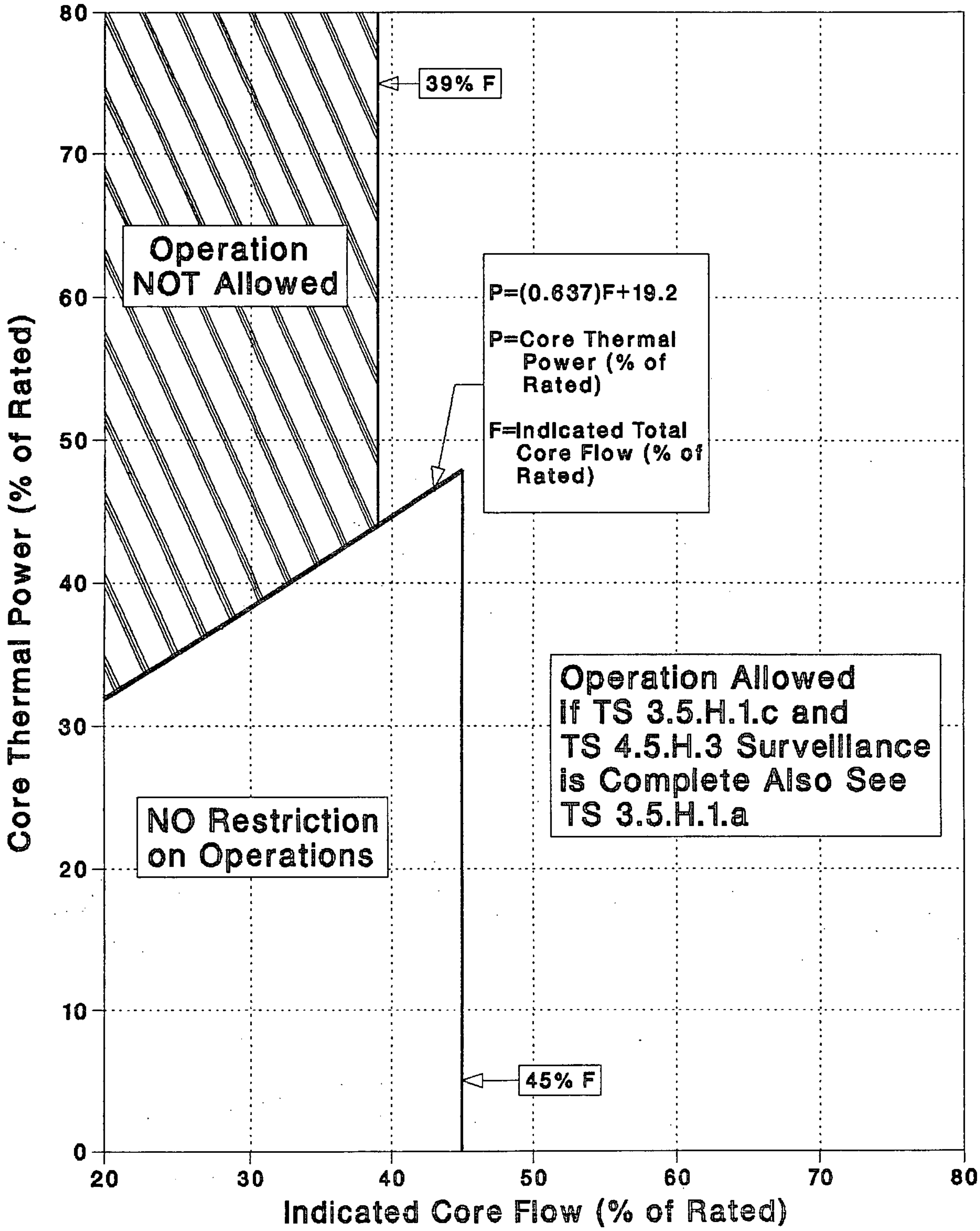

Figure 3.5-1 Single Loop Operation Surveillance Power/Flow Curve

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
C. Minimum Critical Power Ratio (MCPR)	C. Minimum Critical Power Ratio (MCPR)
If thermal power is greater than 45%, the MCPR limit is the greater of: 1.30 1) ~~MCPR (100) from Table 3.11.2~~ multiplied by K_p from Figure 3.11.3 or, 2) $MCPR_F$ from Figure 3.11.4. If thermal power is less than or equal to 45%, the MCPR limit is obtained from Figure 3.11.3. The OLMCPR limit for one recirculation loop operation is 0.01 higher than the comparable two loop value. If at any time during operation it is determined that the limiting value for MCPR is being exceeded, action shall be initiated within 15 minutes to restore operation to within the prescribed limits. Surveillance and corresponding action shall continue until reactor operation is within the prescribed limits. If the steady state MCPR is not returned to within the prescribed limits within two (2) hours, the reactor shall be brought to the Cold Shutdown condition within 36 hours.	MCPR shall be determined daily during reactor power operation at ≥25% rated thermal power and following any change in power level or distribution which has the potential of bringing the core to its operating MCPR Limit.

TABLE 3.11.1
MAXIMUM AVERAGE PLANAR LINEAR HEAT GENERATION RATE vs. EXPOSURE

Exposure	MAPLHGR FOR EACH FUEL TYPE (kw/ft)							
MWD/STU	8DB262 8DB250 8DB219L	8DRB282 8DRB265L	P8DRB265L BP8DRB265L	P8DRB282 BP8DRB282L	P8DRB284LB BP8DRB284LB	P8DRB299L BP8DRB299L	BD 319B	OTHER GE8 FUEL
200	11.1	11.2	11.6	11.2	11.4	11.0	11.19	10.7
1,000	11.3	11.2	11.6	11.2	11.4	11.0	11.31	10.8
5,000	11.9	11.6	11.8	11.8	11.8	11.6	11.99	11.5
10,000	12.0	11.7	11.9	11.9	11.9	11.9	12.60	12.1
15,000	11.9	11.7	11.9	11.8	11.9	11.9	12.34	11.8
20,000	11.8	11.5	11.8	11.7	11.7	11.8	11.95	11.4
25,000	11.3	11.3	11.3	11.3	11.4	11.5	11.56	11.0
30,000	10.2	10.7	10.7	11.1	10.8	10.9	10.54	10.0
35,000	9.6	10.2	10.2	10.4	10.2	10.3	9.53	9.0
40,000	8.9	9.6	9.6	9.8	9.5	9.7	-	-
45,000	-	-	-	-	8.9	9.0	-	-
50,000							6.28	5.8

Note: For two recirculation loop operation. For single loop operation multiply these values by 0.85.

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TABLE 3.11.2

Rated Minimum Critical Power Ratio [MCPR(100)] vs Fuel Type			
Fuel Type	$MCPR_B$ For $\tau ave \leq \tau_B$	MCPR* For τ_B < τave <0.9 SEC	$MCPR_A$ For τave = 0.9 sec
8X8	1.36	*	1.43
P8X8R BP8X8R	1.39	*	1.46

* A linear interpolation between $MCPR_B$ and $MCPR_A$

Bases 3.11

A. Average Planar Linear Heat Generation Rate (APLHGR)

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

The peak cladding temperature 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. Since expected local variations in power distribution within a fuel assembly affect the calculated peak cladding temperature by less than $\pm 20^{\circ}$ 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 at rated conditions conform to 10CFR50.46. The limiting value for APLHGR is given by this specification.

The flow dependent correction factor (Figure 3.11.2) applied to the rated condition's APLHGR limits assures that 1) the 2200°F PCT 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 transients initiated from less than rated core flow conditions. The power dependent correction factor (Figure 3.11.1) applied to the rated conditions APLHGR limits assures that the fuel thermal-mechanical design criteria would be met during abnormal transients initiated from all conditions (Reference 1).

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of LCO. Exceeding APLHGR limits in such cases need not be reported.

B. LHGR

This specification assures that the linear heat generation rate in any rod is less than the design linear heat generation.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of LCO. Exceeding LHGR limits in such cases need not be reported.

C. Minimum Critical Power Ratio (MCPR)

The ECCS evaluation presented in Reference 4 and Reference 6 assumed the steady state MCPR prior to the postulated loss-of-coolant accident to be ~~1.28~~ 1.24 for all fuel types for rated flow. The Rated

Bases Continued

MCPR ~~[MCPR(100)]~~ Limit is determined from the analysis of transients discussed in Bases Sections 2.1 and 2.3. By maintaining an operating MCPR above these limits, the Safety Limit (T.S. 2.1.A) is maintained in the event of the most limiting abnormal operational transient.

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Use of GE's new ODYN code Option B will require average scram time to be a factor in determining the MCPR (Reference 7). In order to increase the operating envelope for MCPR below $MCPR_A$ (ODYN code Option A), the cycle average scram time (τave) must be determined (see Bases 3.3.C). If τave is below the adjusted analysis scram time, the $MCPR_B$ Limit can be used. If $\tau ave > \tau_B$ a linear interpolation must be used to determine the appropriate MCPR. For example:

$$MCPR = MCPR_B + \frac{\tau ave - \tau B}{0.9 - \tau B}(MCPR_A - MCPR_B)$$

$MCPR_A$ and $MCPR_B$ have been determined from the most limiting abnormal operational transients analyses.

The evaluation of a given transient begins with the system initial parameters shown in FSAR Section 14.5 that are input to a GE-core dynamic behavior transient computer program described in References 2 and 3.

At less than 100% of rated flow and power the required MCPR is the larger value of the $MCPR_F$ and $MCPR_P$ at the existing core flow and power state. The required MCPR is a function of flow in order to protect the core from inadvertent core flow increases such that the 99.9% MCPR limit requirement can be assured.

The MCPRs were calculated such that for the maximum core flow rate and the corresponding thermal power along the 105% of rated power/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 MCPRs were calculated at different points along the 105% of rated power flow control line corresponding to different core flows. The calculated MCPR at a given point of core flow ($MCPR_F$) is defined in Figure 3.11.4 (Reference 1).

For operation above 45% of rated thermal power, the core power dependent MCPR operating limit is the rated MCPR limit, MCPR(100), multiplied by the factor, K_P, given in Figure 3.11.3. For operation below 45% of rated thermal power (turbine control valve fast closure and turbine stop valve closure scrams can be bypassed) MCPR limits are established directly from Figure 3.11.3. This protects the core from plant transients other than core flow increase, including a localized event such as rod withdrawal error (Reference 1).

Bases Continued

This limit was determined based upon bounding analyses for the limiting transient at the given core power level. Further information on MCPR operating limits for off-rated conditions is presented in NEDC-30492-P.(1)

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 requirements by a considerable margin. MCPR evaluation below this power level is therefore unnecessary. The daily requirement for calculating MCPR above 25% of rated thermal power is sufficient since power distribution shifts are very slow when there have not been significant power or control rod changes.

Those abnormal operational transients, analyzed in FSAR Section 14.5, which result in an automatic reactor scram are not considered a violation of the LCO. Exceeding MCPR limits in such cases need not be reported.

References

1. "Average Power Range Monitor, Rod Block Monitor and Technical Specification Improvement (ARTS) program for Monticello Nuclear Generating Plant", NEDC-~~30491~~ 30492-P, April, 1984.

— Deleted —

2. ~~"Analytical Methods of Plant Transient Evaluations for the GE BWR", NEDO-10802, February, 1973~~.
3. ~~"Response to NRC Request for Information on ODYN Computer Code", R H Bucholz to P S Check (USNRC), September 28, 1977~~
4. "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K", NEDE-20566, November 1975.
5. "Revision of Low Core Flow Effects on LOCA Analysis for Operating BWRs", R L Gridley (GE) to D G Eisenhut (USNRG), September 28, 1977.
6. "Loss-of-Coolant Accident Analysis Report for the Monticello Nuclear Generating Plant", NEDO-24050-1, December, 1980, L O Mayer (NSP) to Director of Nuclear Reactor Regulation (USNRC), February 6, 1981.
7. "Monticello Nuclear Generating Plant Single-Loop Operation", NEDO-24271, July, 1980

Bases 4.11

The APLHGR, LHGR and MCPR shall be checked daily to determine if fuel burnup, or control rod movement have caused changes in power distribution. Since changes due to burnup are slow, and only a few control rods are removed daily, a daily check of power distribution is adequate. For a limiting value to occur below 25% of rated thermal power, an unreasonably large peaking factor would be required, which is not the case for operating control rod sequences. In addition, the MCPR is checked whenever changes in the core power level or distribution are made which have the potential of bringing the fuel rods to their thermal-hydraulic limits.

5.0 DESIGN FEATURES

5.1 Site

A. The reactor center line is located at approximately 850,810 feet North and 2,038,920 feet East as determined on the Minnesota State Grid, South Zone. The nearest site boundary is approximately 1630 feet S 30° W of the reactor center line and the exclusion area is defined by the minimum fenced area shown in FSAR Figure 2.2.2a. Due to the prevailing wind pattern, the direction of maximum integrated dosage is SSE. The southern property line generally follows the northern boundary of the right-of-way for the Burlington Northern Railway. More details on the current property lines can be found in USAR Figure 2.2-1.

5.2 Reactor

A. The reactor core shall consist of not more than 484 fuel assemblies whose design has been reviewed and approved for BWR use by an NRC Safety Evaluation Report.

B. The reactor core shall contain 121 cruciform-shaped control rods. The control rod material shall be boron carbide powder (B_4C) compacted to approximately 70% of theoretical density~~, except for the Hybrid I control rods which contain approximately 15%~~ or hafnium.

5.3 Reactor Vessel

A. The pressure vessel shall be designed for a pressure of 1250 psig and a temperature of 562°F. The coolant recirculation system shall be designed for a pressure of 1148 psig on suction side of pump and 1248 psig at pump discharge. The applicable design codes shall be as described in Sections 4.2.3 and 4.3.1 of the Monticello Final Safety Analysis Report.

5.4 Containment

A. The primary containment shall be of the pressure suppression type having a drywell and an absorption chamber constructed of steel. The drywell shall have a volume of approximately 134,200 ft^3 and is designed to conform to ASME Boiler and Pressure Vessel Code Section III Class B for an internal pressure of 56 psig at 281°F and an external pressure of 2 psig at 281°F. The absorption chamber shall have a total volume of approximately 176,250 ft^3.

Exhibit C

Monticello Nuclear Generating Plant

License Amendment Request dated July 27, 1987

General Electric Report titled:

Supplemental Reload Licensing Submittal
for
Monticello Nuclear Generating Plant
Cycle 13

23A5827
Revision 0
Class I
June 1987

SUPPLEMENTAL RELOAD LICENSING SUBMITTAL

FOR

MONTICELLO NUCLEAR GENERATING PLANT

CYCLE 13

Prepared: WAZarbis for
P. A. Lambert
Fuel Licensing

Verified: WAZarbis
W. A. Zarbis
Fuel Licensing

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