EXHIBIT A

MONTICELLO NUCLEAR GENERATING PLANT
DOCKET NO. 50-263

1/31/73 - CHANGE REQUEST
PROPOSED CHANGES TO THE TECHNICAL SPECIFICATIONS
APPENDIX A OF PROVISIONAL OPERATING
LICENSE NO. DPR-22

Rod Block Monitor

Page 58, line 4.a of Table 3.2.2:

Change the entry in the column entitled "Run" from "X" to "X(c)."

Page 59, change the second line from the bottom to read:

"c. RBM Upscale and Downscale rod blocks may be bypassed below 30% of rated power."

Page 67, (Bases), insert the following sentence prior to the last sentence in the fifth paragraph:

"This subject is discussed in the General Electric Licensing Topical Report NEDO-10189 70 NED 16 July 1970, An Analysis of Functional Common Mode Failures in GE BWR Protection and Instrumentation."

Reason for change:

The RBM instrumentation is designed such that all trip outputs are bypassed when the reference APRM is below a preset power level. Through an oversight, "Allowable Bypass Condition C" was not worded to include RBM upscale rod blocks. The change in the bases documents the supporting safety evaluation on this subject.

Core and Containment Cooling Systems

Page 108, add the following words to 3.5.G.3:

"except as allowed by Specification 3.5.G.4 below."

Page 108, add 3.5.G.4 reading:

"3.5.G.4 When irradiated fuel is in the reactor vessel and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing or instrument thimble opened at any one time provided that the spent fuel pool gates are open and the fuel pool water level is maintained at a level of greater than or equal to 33 feet."

Page 113, (Bases), change paragraph G to read:

"G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity."

<u>Reason for change</u>:

This requested change will allow control rod drive maintenance concurrent with draining of the suppression chamber for maintenance and inspection. This will safely provide more flexibility in performing maintenance and inspection work during a refueling outage.

A sufficient inventory of water can be maintained such that the ability to cool the core is not jeopardized. The conditions of proposed Specification 3.5.G.4 provide the following safeguards. If only one control rod drive or instrument thimble is removed at one time the largest potential loss of coolant is limited to the approximately two inch diameter opening in the control rod drive nozzle. The instrument thimble involves a much smaller opening. By requiring the fuel pool gate to be open the combined water inventory of the fuel pool, reactor cavity, and dryer/separator pool between the fuel pool low level limit of 33 feet and the reactor vessel flange is 318,000 gallons. When performing

control rod drive maintenance the normal practice after removing a drive is to either immediately insert a rebuilt spare drive or to bolt a flange cover over the drive housing opening. In either event the housing is normally open for less than two hours. For the water inventory available a two inch diameter leak would require over 5 hours to drain the pools to the reactor flange assuming no make-up. At this point the core is still covered with 25 feet of water. Any water draining from the reactor vessel will collect in the suppression chamber; sufficient water will be available for low pressure core cooling system operation before the water level reaches the flange. Therefore, there is adequate time and cooling water available to reestablish core cooling in this unlikely event.

Reactor Recirculation System Crosstie Valve Interlock

Page 108, add items 3.5.I and 4.5.I reading:

"3.5.I Recirculation System

1. Except as specified in 3.5.I.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.

2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.

4.5.I Recirculation System

1. Once per month, when irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be demonstrated to be operable by verifying that the cross tie valves cannot be opened using the normal control switch.

2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily."

Page 113, Bases, add the following:

"I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break.

The peak fuel clad temperature is a function of the size of line break presented in a document entitled, "Monticello Nuclear Generating Plant ECCS Conformance to New AEC Adopted Interim Acceptance Criteria" submitted September 21, 1971. A double ended recirculation line break involves 4.2 sq. ft. when the cross tie valves are closed and 5.6 sq. ft. when the cross tie valves are open. The referenced report shows that the peak fuel clad temperature for the 4.2 sq. ft. rupture while operating at rated power is sufficiently less than the 2300°F limit set forth in the "AEC Adopted Interim Acceptance Criteria for Performance of ECCS for Light-Water Power Reactors" dated June 19, 1971. However, a break of 5.6 sq. ft. will result in clad temperatures in excess of 2300°F. Therefore, at least one cross tie valve must remain closed to reduce the potential break area.

The cross tie valve is allowed to be open during one pump operation. With only one pump, rated power cannot be achieved. Under these conditions, the expected peak clad temperature during a loss of coolant accident is less than that for two pump operation with the cross tie valves closed."

Reason for change:

The ECCS Design Basis as stated in Section 6.2.1.1a of the FSAR states, "No cladding melting shall occur (3370°F)." This was superceded by the AEC Interim ECCS Criteria which states, "The calculated maximum fuel element cladding temperature does not exceed 2300°F." This limit of acceptability is based on the oxidation of the zircaloy cladding to avoid embrittlement and possible fragmentation upon cooldown. General Electric believes there is considerable margin in this limit which should be recognized as being present because a limit of 2700°F appears reasonable for the times of interest to BWR ECCS. Nevertheless, to meet the 2300°F limit set forth, the potential break size restriction requires the cross tie valve to remain closed during two pump operation.

Refueling Interlocks

Page 187, change item 3.10.A to read:

"3.10.A Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alternations and refueling interlocks shall be operable except as specified in Specification 3.10.E."

Reason for change:

The added phrase allows for the addition of Specifications 3.10.E as stated below.

Refueling Interlocks - Extended Core and Control Rod Drive Maintenance

Page 188, add item 3.10.E reading:

"3.10.E Extended Core and Control Rod Drive Maintenance

Control rods may be withdrawn from the reactor core provided the following conditions are satisfied:

1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock input signal from a withdrawn control rod may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable.

2. SRM's shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in 3.10.B."

Page 189, (Bases), insert the following paragraph:

"E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as removal of temporary control curtains, control rod drive maintenance, in-service inspection requirements, examination of the core support plate, etc. When the refueling interlock input signal from a withdrawn control rod is bypassed administrative controls will be in effect to prohibit fuel from being loaded into that control cell.

These operations are performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in Part A of these Bases. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with the control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core."

Reason for change:

This specification is proposed to allow flexibility in performing core and control rod drive maintenance without jeopardizing plant safety. Outage activities will include removal of temporary control curtains, shuffling of fuel, addition of new fuel, in-vessel inspection and control rod drive maintenance. The changes proposed above will permit those activities to be done in a safe, orderly and efficient manner. It should be noted that emptying a complete cell places the reactor in a less reactive condition than with a fully loaded core.

Fuel will not be loaded into a control cell unless the control rod is fully inserted. This is administratively controlled by requiring that whenever such a bypass exists for any control rod in the core, both the operator of the refueling grapple and another licensed reactor operator must verify that a control rod is fully inserted in the control cell prior to loading fuel. In addition, placards will be placed on the core tag board (which is located on the refueling floor for fuel accountability purposes) identifying where fuel loading is not permitted because control rods are withdrawn.

Table 3.2.3 - Continued
Instrumentation That Initiates Rod Block

Function	Trip Settings	Reactor Modes in Which Function Must Be Operable or Operating and Allowable Bypass Conditions**			Total No. of Instrument Channels per Trip System	Min. No. of Operable or Operating Instrument Channels Per Trip System (Notes 1,6)	Required Conditions*
		Refuel	Startup	Run			
4. RBM							
a. Upscale (flow referenced)	≤ .65 + 43 (Note 2)			X(c)	1	1 (Note 5)	D or E
b. Downscale	≥ 2/125 full			X(c)	1	1 (Note 5)	D or E

Notes:

(1) There shall be two operable or operating trip systems for each function. If the minimum number of operable or operating instrument channels cannot be met for one of the two trip systems, this condition may exist up to seven days provided that during this time the operable system is functionally tested immediately and daily thereafter.

(2) "W" is the reactor recirculation driving flow in percent.

(3) Only one of the four SRM channels may be bypassed.

(4) There must be at least one operable or operating IRM channel monitoring each core quadrant

(5) One of the two RBMs may be bypassed for maintenance and/or testing for periods not in excess of 24 hours in any 30 day period. An RBM channel will be considered inoperable if there are less than half the total number of normal inputs from any LPRM level.

EXHIBIT B (Continued)

Table 3.2.3 - Continued

Notes:

(6) Upon discovery that minimum requirements for the number of operable or operating trip systems or instrument channels are not satisfied actions shall be initiated to:

(a) Satisfy the requirements by placing appropriate channels or systems in the tripped condition, or

(b) Place the plant under the specified required conditions using normal operating procedures.

(7) There must be a total of at least 4 operable or operating APRM channels.

*Required conditions when minimum conditions for operation are not satisfied.

A. Reactor in Shutdown mode.

B. No rod withdrawals permitted while in Refuel or Startup mode.

C. Reactor in Run mode.

D. No rod withdrawals permitted while in the Run mode.

E. Power on IRM range or below and reactor in Startup, Refuel, or Shutdown mode.

**Allowable Bypass Conditions

a. SRM Detector-not-fully-inserted rod block may be bypassed when the SRM channel count rate is 100 cps or when all IRM range switches are above Position 2.

b. IRM Downscale rod block may be bypassed when the IRM range switch is in the lowest range position.

c. RBM UPSCALE AND Downscale rod ~~block~~ BLOCKS may be bypassed below 30% rated power.

d. SRM Upscale block may be bypassed when associated IRM range switches are above Position 7.

Bases Continued:

This subject is discussed in the General Electric licensing Topical Report NEDO-10189 70 NED 16 July 1970, An Analysis of Functional Common Mode Failures in GE BWR Protection and Instrumentation.

3.2 The HPCI and/or RCIC high flow and temperature instrumentation is provided to detect a break in the HPCI and/or RCIC piping. Tripping of this instrumentation results in actuation of HPCI and/or RCIC isolation valves; i.e., Group 4 and/or Group 5 valves. The trip settings of 200°F and 150% of design flow and valve closure time are such that core uncovery is prevented and fission product release is within 10 CFR 100 guidelines.

The instrumentation which initiates ECCS action is arranged in a dual bus system. As for other vital instrumentation arranged in this fashion the Specification preserves the effectiveness of the system even during periods when maintenance or testing is being performed.

The control rod block functions are provided to prevent excessive control rod withdrawal so that MCHFR does not decrease to 1.0. The trip logic for this function is 1 out of n; e.g., any trip on one of the six APRM's, eight IRM's, or four SRM's will result in a rod block. The minimum instrument channel requirements for the IRM and RBM may be reduced by one for a short period of time to allow for maintenance, testing, or calibration. See Section 7.3 FSAR.

The APRM rod block trip is referenced to flow and prevents a signigicant reduction in MCHFR especially during operation at reduced flow. The APRM provides gross core protection; i.e., limits the gross core power increase from withdrawal of control rods in the normal withdrawal sequence. The trips are set so that MCHFR is maintained greater than 1.0.

THE RBM provides local protection of the core; i.e., the prevention of critical heat flux 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 flow. The worst case single control rod withdrawal error has been analyzed and the results show that with the specified trip settings rod withdrawal is blocked when MCHFR is ~1.3, thus allowing adequate margin. Ref. Section 7.4.5.3 and 14.5.3 FSAR. Below 70% power the worst case withdrawal of a single control rod results in a MCHFR >1.0 without rod block action, thus below this level it is not required. Requiring at least half of the normal LPRM inputs from each level to be operable assures that the RBM response will be adequate to prevent rod withdrawal errors.

The IRM rod block function provides local as well as gross core protection. The scaling arrangement is such that trip setting is less than a factor of 10 above the indicated level. Analysis of the worst case accident results in rod block action before MCHFR approaches 1.0. Ref. Section 7.4.4.3 FSAR.

A downscale indication of an APRM or IRM is an indication the instrument has failed or the instrument is not sensitive enough. In either case the instrument will not respond to changes in control rod motion and thus control rod motion is prevented. The downscale trips are set at 3/125 of full scale.

EXHIBIT B (Continued)

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
3. When irradiated fuel is in the reactor vessel and reactor coolant temperature is less than 212°F, all low pressure core and containment cooling subsystems may be inoperable provided no work is being done which has the potential for draining the reactor vessel, EXCEPT AS ALLOWED BY SPECIFICATION 3.5.G.4 BELOW.	4. When irradiated fuel is in the reactor vessel and the vessel head is removed, the suppression chamber may be drained completely and no more than one control rod drive housing or instrument thimble opened at any one time provided that the spent fuel pool gates are open and the fuel pool water level is maintained at a level of greater than or equal to 33 feet.
H. Extended Maintenance When it is determined that maintenance to restore components or systems to an operable condition will last longer than the periods specified, a report detailing the circumstances and the estimated date for returning the components or systems to an operable condition shall be submitted to the AEC prior to the allowable end of the out-of-service period.	
I. Recirculation System	I. Recirculation System
1. Except as specified in 3.5.I.2 below, whenever irradiated fuel is in the reactor, with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be operable.	1. Once per month, when irradiated fuel is in the reactor with reactor coolant temperature greater than 212°F and both reactor recirculation pumps operating, the recirculation system cross tie valve interlocks shall be demonstrated to be operable by verifying that the cross tie valves cannot be opened using the normal control switch.
2. The recirculation system cross tie valve interlocks may be inoperable if at least one cross tie valve is maintained fully closed.	2. When a recirculation system cross tie valve interlock is inoperable, the position of at least one fully closed cross tie valve shall be recorded daily.

Bases Continued 3.5:

G. Emergency Cooling Availability

INSERT ON FOLLOWING PAGE REPLACES PARAGRAPH G

~~The purpose of Specification G is to assure that at least a minimum of core cooling equipment is avail-~~ able at all times. It is during refueling outages that major maintenance is performed and during such time that all low pressure core cooling systems may be out of service. This specification provides that should this occur, no work will be performed on the primary system which could lead to draining the vessel. This work would include work on certain control rod drive components and recirculation ~~system. Thus, the specification precludes the events which could require core cooling.~~

H. Extended Maintenance

Nearly all maintenance can be completed within a few days. Infrequently, however, major maintenance might be required. Replacement of principal system components could necessitate outages of more than the time allowed for a system or component to be out of service. In spite of the best efforts of the operator to return equipment to service, some maintenance could require up to six months.

I. Recirculation System

The capacity of the Emergency Core Coolant System is based on the potential consequences of a double ended recirculation line break. The peak fuel clad temperature is a function of the size of line break presented in a document entitled, "Monticello Nuclear Generating Plant ECCS Conformance to New AEC Adopted Interim Acceptance Criteria" submitted September 21, 1971. A double ended recirculation line break involves 4.2 sq. ft. when the cross tie valves are closed and 5.6 sq. ft. when the cross tie valves are open. The referenced report shows that the peak fuel clad temperature for the 4.2 sq. ft. rupture while operating at rated power is sufficiently less than the 2300°F limit set forth in the "AEC Adopted Interim Acceptance Criteria for Performance of ECCS for Light-Water Power Reactors" dated June 19, 1971. However, a break of 5.6 sq. ft will result in clad temperatures in excess of 2300°F. Therefore, at least one cross tie valve must remain closed to reduce the potential break area.

The cross tie valve is allowed to be open during one pump operation. With only one pump, rated power cannot be achieved. Under these conditions, the expected peak clad temperature during a loss of coolant accident is less than that for two pump operation with the cross tie valve closed.

G. Emergency Cooling Availability

The purpose of Specification G is to assure that sufficient core cooling equipment is available at all times. It is during refueling outages that major maintenance is performed and during such time that all core and containment cooling subsystems may be out of service. Specification 3.5.G.3 allows all core and containment cooling subsystems to be inoperable provided no work is being done which has the potential for draining the reactor vessel. Thus events requiring core cooling are precluded.

Specification 3.5.G.4 recognizes that concurrent with control rod drive maintenance during the refueling outage, it may be necessary to drain the suppression chamber for maintenance or for the inspection required by Specification 4.7.A.1. In this situation, a sufficient inventory of water is maintained to assure adequate core cooling in the unlikely event of loss of control rod drive housing or instrument thimble seal integrity.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.10 REFUELING

Applicability:

Applies to fuel handling and core reactivity limitations.

Objective:

To assure core reactivity is within capability of the control rods and to prevent criticality during refueling.

Specification:

A. Refueling Interlocks

The reactor mode switch shall be locked in the "Refuel" position during core alterations and the refueling interlocks shall be operable, EXCEPT AS SPECIFIED IN SPECIFICATION 3.10.E.

4.10 REFUELING

Applicability:

Applies to the periodic testing of those interlocks and instruments used during refueling.

Objective:

To verify the operability of instrumentation and interlocks used in refueling.

Specification:

A. Refueling Interlocks

Prior to any fuel handling, with the head off the reactor vessel, the refueling interlocks shall be functionally tested. They shall also be tested at weekly intervals thereafter until no longer required and following any repair work associated with the interlocks.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS

3.0 LIMITING CONDITIONS FOR OPERATION

B. Core Monitoring

During core alterations two SRM's shall be operable, one in and one adjacent to any core quadrant where fuel or control rods are being moved. For an SRM to be considered operable, the following conditions shall be satisfied:

1. The SRM shall be inserted to the normal operating level. (Use of special moveable, dunking type detectors during initial fuel loading and major core alterations is permissible as long as the detector is connected into the normal SRM circuit.)

2. The SRM shall have a minimum of 3 CPS with all rods fully inserted in the core.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool, the pool water level shall be maintained at a level of greater or equal to 33 feet.

D. The reactor shall be shutdown for a minimum of 24 hours prior to movement of fuel within the reactor.

NOTE: ADD PAGE 188a

4.0 SURVEILLANCE REQUIREMENTS

B. Core Monitoring

Prior to making any alterations to the core, the SRM's shall be functionally tested and checked for neutron response. Thereafter, the SRM's will be checked daily for response.

C. Fuel Storage Pool Water Level

Whenever irradiated fuel is stored in the fuel storage pool the pool level shall be recorded daily.

3.0 LIMITING CONDITIONS FOR OPERATION	4.0 SURVEILLANCE REQUIREMENTS
E. Extended Core and Control Rod Drive Maintenance Control rods may be withdrawn from the reactor core provided the following conditions are satisfied: 1. The reactor mode switch shall be locked in the "Refuel" position. The refueling interlock input signal from a withdrawn control rod may be bypassed on a withdrawn control rod after the fuel assemblies in the cell containing (controlled by) that control rod have been removed from the reactor core. All other refueling interlocks shall be operable. 2. SRM's shall be operable in the core quadrant where fuel or control rods are being moved, and in an adjacent quadrant. The requirements for an SRM to be considered operable are given in 3.10.B.	

Bases Continued:

E. The intent of this specification is to permit the unloading of a significant portion of the reactor core for such purposes as removal of temporary control curtains, control rod drive maintenance, in-service inspection requirements, examination of the core support plate, etc. When the refueling interlock input signal from a withdrawn control rod is bypassed, administrative controls will be in effect to prohibit fuel from being loaded into that control cell.

These operations are performed with the mode switch in the "Refuel" position to provide the refueling interlocks normally available during refueling as explained in Part A of these Bases. In order to withdraw more than one control rod, it is necessary to bypass the refueling interlock on each withdrawn control rod. The requirement that the fuel assemblies in the cell controlled by the control rod be removed from the reactor core before the interlock can be bypassed insures that withdrawal of another control rod does not result in inadvertent criticality. Each control rod essentially provides reactivity control for the fuel assemblies in the cell associated with the control rod. Thus, removal of an entire cell (fuel assemblies plus control rod) results in a lower reactivity potential of the core.

Bases:

A. Refueling Interlocks

During refueling operations, the reactivity potential of the core is being altered. It is necessary to require certain interlocks and restrict certain refueling procedures such that there is assurance that inadvertent criticality does not occur.

To minimize the possibility of loading fuel into a cell containing no control rod, it is required that all control rods are fully inserted when fuel is being loaded into the reactor core. This requirement assures that during refueling the refueling interlocks, as designed, will prevent inadvertent criticality. The core reactivity limitation of Specification 3.3 limits the core alterations to assure that the resulting core loading can be controlled with the reactivity control system and interlocks at any time during shutdown or the following operating cycle.

Addition of large amounts of reactivity to the core is prevented by operating procedures, which are in turn backed up by refueling interlocks on rod withdrawal and movement of the refueling platform. When the mode switch is in the "Refuel" position, interlocks prevent the refueling platform from being moved over the core if a control rod is withdrawn and fuel is on a hoist. Likewise, if the refueling platform is over the core with fuel on a hoist, control rod motion is blocked by the interlocks. With the mode switch in the refuel position only one control rod can be withdrawn.

For a new core the dropping of a fuel assembly into a vacant fuel location adjacent to a withdrawn control rod does not result in an excursion or a critical configuration, thus adequate margin is provided.

B. Core Monitoring

The SRM's are provided to monitor the core during periods of station shutdown and to guide the operator during refueling operations and station startup. Requiring two operable SRM's, one in and one adjacent to any core quadrant where fuel or control rods are being moved, assures adequate monitoring of that quadrant during such alterations. The requirement of 3 counts per second provides assurance that neutron flux is being monitored.

C. Fuel Storage Pool Water Level

To assure that there is adequate water to shield and cool the irradiated fuel assemblies stored in the pool, a minimum pool water level is established. The minimum water level of 33 feet is established because it would be a significant change from the normal level (37'9") and well above a level to assure adequate cooling.

NOTE: ADD PAGE 189a