EXHIBIT A

Monticello Nuclear Generating Plant

License Amendment Request Dated March 30, 1984

Miscellaneous Technical Specification Changes

Proposed Changes to the Technical Specifications,
Appendix A of Operating License DPR-22

Pursuant to 10 CFR Part 50, Section 50.59 and Section 50.90, the holders of Operating License DPR-22 hereby propose the following changes to Appendix A, Technical Specifications.

1. Security Plan Implementing Procedures Review

Proposed Changes
Drop Operation Committee review of non-safety related procedures governing work activities exclusively applicable to or performed by the guards as shown in Exhibit B pages 242, 244 and 246b.

Reason for Changes
The Operations Committee should not be required to review non-safety related procedures written to cover details of guard force functions.

Significant Hazards Evaluation
Operations Committee review of security procedures will be omitted only for non-safety related procedures associated with activities performed exclusively by security personnel.

The proposed change is a purely administrative change to the Technical Specifications. For these reasons operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

2. Intake Structure Sprinkler System

Proposed Change
Add the intake structure sprinkler system to the Technical Specifications as shown in Exhibit B page 227a.

Reason for Change
The Appendix R, Intake Structure Sprinkler System modification is complete. This change adds the intake structure sprinkler system to the Technical Specifications.

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Significant Hazards Evaluation

The proposed change adds limiting conditions for operation and surveillance requirements for the intake structure sprinkler system constituting additional limitations, restrictions and controls not presently included in the Technical Specifications.

For these reasons operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

3. Page Number Reference Update and Miscellaneous Typographical Errors

Proposed Changes

Update the Technical Specifications as shown in Exhibit B, pages 4, 17, 18, 19, 20, 84, 117, 151, 172, 201, 209, 227a and 229p.

Reason for Changes

Updating the Technical Specifications will change as follows:

Page 17	-	A page number reference is corrected.
Page 18	-	Two page number references are corrected and a punctuation mark is corrected.
Page 19	-	A page number reference is corrected.
Page 20	-	A page number reference is corrected.
Page 84	-	A sentence is clarified with respect to reactivity margin. The phrase "... in the most reactive condition during the operating cycle ..." at the beginning of the cycle and the phrase, "... the initial loading ..." is replaced with "... at the beginning of the cycle ..." The existing wording is misleading and in error if strictly interpreted.
Page 117	-	Change "facts" to "fact". This is á typographical error.
Page 201	-	Change "... shall be demonstrated to be operable at least once each day ..." to "... shall be demonstrated to be operable immediately and daily thereafter." This is con-consistent with other Technical Specification requirements and is a more conservative action.
Page 209a	-	Add a bases statement for 3.10.D The proposed language simply states the FSAR basis for the 24-hour shutdown period requirement for fuel movement. Each Limiting Condition for Operation should be supported by its bases.
Page 227a	-	Correct spelling of "Lube" and "suppression".
Page 229p	-	Correct the specified fish LLD for Fe-59. The value of 260 is consistent with the requirements of NUREG-0472. This is a typographical error correction.

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Significant Hazards Evaluation
The proposed changes to the Technical Specifications are purely administrative in nature.

For these reasons operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

4. Schedule for Containment Integrated Leak Rate Test

Proposed Changes
Revise the footnote on page 157 of the Technical Specifications to permit a one-time deviation in the specified Type A overall integrated containment leakage rate test for the 1984 refueling outage. The following addition is proposed:

> ... The first test of the second 10-year period shall be conducted during the 1984 refueling shutdown.

Reason for Change
The Technical Specifications require a test interval of 40 $\pm$ 10 months for the overall integrated containment leakage rate test. The last test was completed on May 8, 1980. The next test must be performed by July 8, 1984 to meet this requirement. Due to the length of the current refueling outage, which has been extended to mid-October, 1984 to accomodate replacement of the recirculation system piping, the test cannot be completed as required.

Because of the number of plant maintenance and modification projects in progress, including containment modifications as part of the Mark I containment long-term program, the containment integrated leakage rate test should be scheduled at the end of the outage following all major work. This will provide assurance of the integrity of the containment vessel following this period of extensive maintenance and modification. A September or October test date is therefore required and a one-time deviation from the 40 $\pm$ 10 month schedule is needed. Performance of this test at the beginning or midway through the outage would serve no practical purpose and would severely impact the outage schedule and add to the occupational radiation exposure incurred.

Significant Hazards Evaluation
The proposed change would extend the allowable interval between overall integrated containment leakage rate tests by approximately two or three months. The exact extension is unknown since the outage may be extended (or shortened) based on progress of the modification and maintenance projects underway. This extension will permit the test to be performed at the end of the outage providing assurance of containment integrity.

We believe the requested change does not involve a significant hazards consideration since the change would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) involve a significant reduction in a margin of safety.

5. Clarification of Main Steam Line Isolation Valves Operating Time

Proposed Change
Clarify the main steam line isolation valves operating times as shown in Exhibit B page 172.

Reason for Change
To clarify the main steam line isolation valves operating times.

Significant Hazards Evaluation
The proposed change in Table 3.7.1 clarifies the main steam line isolation valves operating times. Thus the proposed change is a purely administrative change to the Technical Specifications.

For these reasons operation of the Monticello Nuclear Generating Plant in accordance with the proposed changes would not:

(1) Involve a significant increase in the probability or consequences of an accident previously evaluated; or

(2) Create the possibility of a new or different kind of accident from any accident previously evaluated; or

(3) Involve a significant reduction in a margin of safety.

EXHIBIT B

License Amendment Request dated March 30, 1984

Docket No. 50-263 License No. DPR-22

Exhibit B consists of revised pages for the Monticello Nuclear Generating Plant Technical Specifications as listed below:

Pages

17
18
19
20
84
117
157
172
201
209a
227a
229p
242
244
246b

backed up by the rod worth minimizer. Worth of individual rods is very low in a uniform rod pattern. Thus, of all possible sources of reactivity input, uniform control rod withdrawal is the most probable cause of significant power rise. Because the flux distribution associated with uniform rod withdrawals does not involve high local peaks, and because several rods must be moved to change power by a significant percentage of rated power, the rate of power rise is very slow. Generally, the heat flux is in near equilibrium with the fission rate. In an assumed uniform rod withdrawal approach to the scram level, the rate of power rise is no more than 5% of rated power per minute, and the IRM system would be more than adequate to assure a scram before the power could exceed the safety limit. The IRM scram remains active until the mode switch is placed in the run position. This switch occurs when reactor pressure is greater than 850 psig.

The analysis to support operation at various power and flow relationships has considered operation with either one or two recirculation pumps. During steady-state operation with one recirculation pump operating the equalizer line shall be open. Analysis of transients from this operating condition are less severe than the same transients from the two pump operation.

The operator will set the APRM neutron flux trip setting no greater than that stated in Specification 2.3.A.1. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.A.1 for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

B. APRM Control Rod Block Trips Reactor power level may be varied by moving control rods or by varying the recirculation flow rate. The APRM system provides a control rod block to prevent rod withdrawal beyond a given point at constant recirculation flow rate, and thus to protect against the condition of a MCPR less than the Safety Limit (T.S.2.1.A). This rod block trip setting, which is automatically varied with recirculation loop flow rate, prevents an increase in the reactor power level to excessive values due to control rod withdrawal. The flow variable trip setting provides substantial margin from fuel damage, assuming a steady-state operation at the trip setting, over the entire recirculation flow range. The margin to the Safety Limit increases as the flow decreases for the specified trip setting versus flow relationship; therefore, the worst case MCPR which could occur during steady-state operation is at 108% of rated thermal power because of the APRM rod block trip setting. The actual power distribution in the core is established by specified control rod sequences and is monitored by the in-core LPRM system. When the maximum fraction of limiting power density exceeds the fraction of rated thermal reactor power, the rod block setting is adjusted in accordance with the formula in Specification 2.3.B. If the APRM rod block setting should require a change due to an abnormal peaking condition, it will be done by increasing the APRM gain and thus reducing the slope and intercept point of the flow referenced rod block curve by the reciprocal of the APRM gain change.

The operator will set the APRM rod block trip settings no greater than that stated in Specification 2.3.B. However, the actual setpoint can be as much as 3% greater than that stated in Specification 2.3.B for recirculation driving flows less than 50% of design and 2% greater than that shown for recirculation driving flows greater than 50% of design due to the deviations discussed on page 39.

C. Reactor Low Water Level Scram The reactor low water level scram is set at a point which will assure that the water level used in the bases for the safety limit is maintained.

The operator will set the low water level trip setting no lower than 10'6" above the top of the active fuel. However, the actual setpoint can be as much as 6 inches lower due to the deviations discussed on page 39.

D. Reactor Low Low Water Level ECCS Initiation Trip Point The emergency core cooling subsystems are designed to provide sufficient cooling to the core to dissipate the energy associated with the loss of coolant accident and to limit fuel clad temperature to well below the clad melting temperature to assure that core geometry remains intact and to limit any clad metal-water reaction to less than 1%. The design of the ECCS components to meet the above criterion was dependent on three previously set parameters; the maximum break size, the low water level scram setpoint, and the ECCS initiation setpoint. To lower the setpoint for initiation of the ECCS could prevent the ECCS components from meeting their criterion. To raise the ECCS initiation setpoint would be in a safe direction, but it would reduce the margin established to prevent actuation of the ECCS during normal operation or during normally expected transients.

The operator will set the low low water level ECCS initiation trip setting $\geq$ 6'6" $\leq$ 6'10" above the top of the active fuel. However, the actual setpoint can be as much as 3 inches lower than the 6'6" setpoint and 3 inches greater than the 6'10" setpoint due to the deviations discussed on page 39.

E. Turbine Control Valve Fast Closure Scram The turbine control valve fast closure scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to a load rejection and subsequent failure of the bypass. This transient is less severe than the turbine stop valve closure with bypass failure and therefore adequate margin exists.

F. Turbine Stop Valve Scram The turbine stop valve closure scram trip anticipates the pressure, neutron flux and heat flux increase that could result from rapid closure of the turbine stop valves. With a scram trip setting of $\leq$ 10% of valve closure from full open, the resultant increase in surface heat flux is limited such that MCPR remains above the Safety Limit (T.S.2.1.A) even during the worst case transient that assumes the turbine bypass is closed.

G. Main Steam Line Isolation Valve Closure Scram The main steam line isolation valve closure scram anticipates the pressure and flux transients which occur during normal or inadvertent isolation closure. With the scram set at 10% valve closure there is no increase in neutron flux.

H. Main Steam Line Low Pressure Initiates Main Steam Isolation Valve Closure The low pressure isolation of the main steam lines at 825 psig was provided to give protection against rapid reactor depressurization and the resulting rapid cooldown of the vessel. Advantage was taken of the scram feature which occurs when the main steam line isolation valves are closed to provide for reactor shutdown so that high power operation at low reactor pressure does not occur, thus providing protection for the fuel cladding integrity safety limit. Operation at steamline pressures lower than 825 psig requires

Bases Continued:

that the reactor mode switch be in the startup position where protection of the fuel cladding integrity safety limit is provided by the IRM high neutron flux scram. Thus, the combination of main steam line low pressure isolation and isolation valve closure scram assures the availability of the neutron scram protection over the entire range of applicability of the fuel cladding integrity safety limit.

The operator will set this pressure trip at greater than or equal to 825 psig. However, the actual trip setting can be as much as 10 psi lower due to the deviations discussed on page 39.

References

1. Linford, R. B., "Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor", NEDO-10802, Feb., 1973.

Bases Continued 3.3 and 4.3:

A. Reactivity Limitations

1. Reactivity Margin - core loading

The core reactivity limitation is a restriction to be applied principally to the design of new fuel which may be loaded in the core or into a particular refueling pattern. Satisfaction of the limitation can only be demonstrated at the time of loading and must be such that it will apply to the entire subsequent fuel cycle. The generalized form is that the reactivity of the core loading will be limited so the core can be made subcritical by at least R + 0.25% Δk at the beginning of the cycle, with the strongest control rod fully withdrawn and all others fully inserted. The value of R in % Δk is the amount by which the core reactivity, at any time in the operating cycle, is calculated to be greater than at the time of the check; i.e., at the beginning of the cycle. R must be a positive quantity or zero. A core which contains temporary control or other burnable neutron absorbers may have a reactivity characteristic which increases with core lifetime, goes through a maximum and then decreases thereafter. See Figure 3.3.2 of the FSAR for such a curve.

The value of R is the difference between the calculated core reactivity at the beginning of the operating cycle and the calculated value of core reactivity any time later in the cycle where it would be greater than at the beginning. The value of R shall include the potential shutdown margin loss assuming full B_4C settling in all inverted poison tubes present in the core. New values of R must be calcualted for each new fuel cycle.

The 0.25% Δk in the expression R + 0.25% Δk is provided as a finite, demonstrable, sub-criticality margin. This margin is demonstrated by full withdrawal of the strongest rod and partial withdrawal of an adjacent rod to a position calculated to insert at least R + 0.25% Δk in reactivity. Observation of sub-criticality in this condition assures sub-criticality with not only the strongest rod fully withdrawn but at least a R + 0.25% Δk margin beyond this.

2. Reactivity margin - stuck control rods

Specification 3.3.A.2 requires that a rod be taken out of service if it cannot be moved

C. RHR Service Water

The containment heat removal portion of the RHR system is provided to remove heat energy from the containment in the event of a loss of coolant accident. For the flow specified, the containment longterm pressure is limited to less than 5 psig and, therefore, is more than ample to provide the required heat removal capability. Reference Section 6.2.3.2.3. FSAR. The repair periods specified were arrived at as in 3.5.B above.

The containment cooling subsystem consists of two sets of 2 service water pumps, 1 heat exchanger, and 2 RHR pumps. Either set of equipment is capable of performing the containment cooling function. Loss of one RHR service water pump does not seriously jeopardize the containment cooling capability as two of the remaining three pumps can satisfy the cooling requirements. Since there is some redundancy left, a 30 day repair period is adequate. Loss of 1 containment cooling subsystem leaves one remaining system to perform the containment cooling function. The operable system is demonstrated to be operable each day when the above condition occurs. Based on the fact that when one containment cooling subsystem becomes inoperable only one system remains which is tested daily. A 7 day repair period was specified.

The RHR service water system provides cooling for the RHR heat exchangers and can thus maintain the suppression pool water within limits. With the flow specified, the pool temperature limits are maintained as specified in Specification 3.7.A.1.

D. High Pressure Coolant Injection

The high pressure coolant injection system is provided to adequately cool the core for all pipe breaks smaller than those for which the LPCI or core spray subsystems can protect the core.

The HPCI meets this requirement without the use of off-site AC power. For the pipe breaks for which the HPCI is intended to function, the core never uncovers and is continuously cooled and thus no clad damage occurs. Reference Section 6.2.4.3 FSAR.

The HPCI system is backed up by the automatic pressure relief system and either of two core spray systems or the LPCI system. Therefore, when the HPCI system is out of service, the automatic pressure relief and core spray systems and LPCI system are required to be operable. For additional

d. During reactor isolation conditions the reactor pressure vessel shall be depressurized to <200 psig at normal cooldown rates if the suppression pool temperature exceeds 120°F.

e. The suppression chamber water volume shall be $\geq$68,000 and $\leq$77,970 cubic feet.

f. Two channels of torus water level instrumentation shall be operable. From and after the date that one channel is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding 30 days unless such channel is sooner made operable. If both channels are made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding six hours unless at least one channel is sooner made operable.

2. Primary Containment Integrity

Primary containment integrity, as defined in Section 1, shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F and fuel is in the reactor vessel except while performing low power physics tests at atmospheric pressure during or after refueling at power levels not to exceed 5 Mw(t).

d. Whenever there is indication of relief valve operation with a suppression pool temperature $\geq$160°F and the primary coolant system pressure >200 psig, an extended visual examination of the suppression chamber shall be conducted before resuming power operation.

e. The suppression chamber water volume shall be checked once per day.

f. The suppression chamber water volume indicators shall be calibrated semi-annually.

2. Primary Containment Integrity

a. Integrated Primary Containment Leak Test (IPCLT)

The containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR 50 using the methods and provisions of ANSI N45.4-1972:

1. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 $\pm$ 10 month intervals during shutdown at $\overline{P}_a$ (41 psig) during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.*

*The third test of the first 10-year service period shall be conducted during the 1980 refueling shutdown. The first test of the second 10-year period shall be conducted during the 1984 refueling shutdown.

TABLE 3.7.1

PRIMARY CONTAINMENT ISOLATION

Isolation Group	Valve Identification	Number of Valves		Maximum Operating Time (Sec)	Normal Position
		Inboard	Outboard		
1	Main Steam Line Isolation	4	4	5*	Open
1	Main Steam Line Drain	1	1	60	Closed
1	Recirculation Loop Sample Line	1	1	60	Closed
2	Drywell Floor Drain		2	60	Open
2	Drywell Equipment Drain		2	60	Open
2	Drywell Vent		2	60	Closed
2	Drywell Vent Bypass		1	60	Closed
2	Drywell Purge Inlet		2	60	Open
2	Drywell and Suppression Chamber Air Makeup		1	60	Closed
2	Suppression Chamber to Drywell N_2 Recirculation		1	60	Open
2	Suppression Chamber Vent		2	60	Closed
2	Suppression Chamber Vent Bypass		1	60	Open
2	Shutdown Cooling System	1	1	120	Closed

*Minimum closure time shall be $\geq$3 seconds service providing both the emergency diesel generators are operable.

2. Reserve Transformers

During power operation one reserve transformer may be out of service for maintenance if the second reserve transformer is operational and available for automatic operation on loss of normal auxiliary power.

3. Standby Diesel Generators

a. From and after the date that one of the diesel generators is made or found to be inoperable for any reason, reactor operation is permissible only during the succeeding seven days unless such diesel generator is sooner made operable, provided that during such seven days the operable diesel generator shall be demonstrated to be operable immediately and daily thereafter.

b. If both diesel generators become inoperable during power operation, the reactor shall be placed in the cold shutdown condition.

B. 3. Standby Diesel Generators

a. Each diesel generator shall be manually started and loaded once every month to demonstrate operational readiness. The test shall continue until both the diesel engine and the generator are at equilibrium conditions of temperature while full load output is maintained.

b. During the monthly generator test, the diesel starting air compressor shall be checked for operation and their ability to recharge air receivers.

Bases (continued):

D. Minimum Shutdown Period

A minimum shutdown period of 24 hours is specified prior to movement of fuel within the reactor since analysis of refueling accidents assume a 24-hour decay time following extended operation at power. Since the reactor must be shut down, depressurized, and the head removed prior to moving fuel, it is not expected that fuel could actually be moved in less than 24 hours.

3.0 LIMITING CONDITIONS FOR OPERATION

E. Sprinkler Systems

1. The following spray or sprinkler systems shall be operable whenever equipment in the protected area(s) is required to be operable:

   a. Diesel Generator and Day Tank Rooms
   b. Lube Oil Drum Storage
   c. Lube Oil Storage Tank Sprinkler
   d. Hydrogen Seal Oil Unit Sprinkler
   e. Lube Oil Piping System Sprinkler
   f. Lube Oil Reservoir
   g. Recirc MG Set Sprinklers
   h. Intake Structure

2. If Specification 3.13.E.1 cannot be met, within one hour establish a continuous fire watch with backup fire suppression equipment for the unprotected area(s). Restore the system to operable status within 14 days or submit a 30-day written report outlining the cause of the inoperability and the plans and schedule for restoring the system to operable status.

4.0 SURVEILLANCE REQUIREMENTS

E. Sprinkler Systems

1. Each of the spray or sprinkler systems listed in specification 3.13.E.1 shall be demonstrated operable as follows:

   a. Each valve (manual, power operated, or automatic) in the flow path that is not electrically supervised, locked, sealed or otherwise secured in position, shall be verified to be in its correct position every month.

   b. Cycle each testable valve in the flow path through at least one complete cycle of full travel once each year.

   c. Perform a system functional test every 18 months which includes, where applicable, simulated automatic actuation of the system and verification that the automatic valves in the flow path actuate to their correct positions on a test signal.

   d. At least once per 5 years by performing an air flow test through each open head sprinkler header and verifying each open head sprinkler is unobstructed.

   e. At least once per 18 months by a visual examination of system piping and sprinkler heads. An air flow test shall be performed upon evidence of obstruction of any open head sprinkler.

Table 4.16.2
(Page 1 of 2)

MAXIMUM VALUES FOR THE LOWER LIMITS OF DETECTION (LLD)[a,e]

Analysis	Water (pCi/l)	Airborne Particulate or Gas (pCi/m^3)	Fish (pCi/kg, wet)	Milk (pCi/l)	Food Products (pCi/kg, wet)	Sediment (pCi/kg, dry)
gross beta	4[b]	1×10^{-2}				
^{3}H	2000(1000[b])					
^{54}Mn	15		130			
^{59}Fe	30		260			
$^{58,60}Co$	15		130			
^{65}Zn	30		260			
^{95}Zr-Nb	15[c]					
^{131}I	1[b, d]	7×10^{-2}		1[d]	60	
$^{134,137}Cs$	15(10[b]), 18	1×10^{-2}	130	15	60	150
^{140}Ba-La	15[c]			15[c]		

f. All events which are required by regulation or technical specifications to be reported to NRC in writing within 24 hours.

g. Drills on emergency procedures (including plant evacuation) and adequacy of communication with off-site support groups.

h. All procedures required by these Technical Specifications, including implementing procedures of the Emergency Plan and the Security Plan (except as exempted in Section 6.5.F), shall be reviewed with a frequency commensurate with their safety significance but at an interval of not more than two years.

i. Perform special reviews and investigations, as requested by the Safety Audit Committee.

j. Review of investigative reports of unplanned releases of radioactive material to the environs.

k. All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM).

5. Authority

The OC shall be advisory to the Plant Manager. In the event of disagreement between the recommendations of the OC and the Plant Manager, the course determined by the Plant Manager to be the more conservative will be followed. A written summary of the disagreement will be sent to the General Manager Nuclear Plants and the Chairman of the SAC for review.

6. Records

Minutes shall be recorded for all meetings of the OC and shall identify all documentary material reviewed. The minutes shall be distributed to each member of the OC, the Chairman and each member of the Safety Audit Committee, the General Manager Nuclear Plants and others designated by OC Chairman or Vice Chairman.

7. Procedures

A written charter for the OC shall be prepared that contains:

a. Responsibility and authority of the group.

b. Content and method of submission of presentations to the Operations Committee.

6.5 Plant Operating Procedures

Detailed written procedures, including the applicable check-off lists and instructions, covering areas listed below shall be prepared and followed. These procedures and changes thereto, except as specified in 6.5.G shall be reviewed by the Operation Committee and approved by a member of plant management designated by the Plant Manager.

A. Plant Operations

1. Integrated and system procedures for normal startup, operation and shutdown of the reactor and all systems and components involving nuclear safety of the facility.

2. Fuel handling operations.

3. Actions to be taken to correct specific and foreseen potential or actual malfunction of systems or components including responses to alarms, primary system leaks and abnormal reactivity changes and including follow-up actions required after plant protective system actions have initiated.

4. Surveillance and testing requirements that could have an effect on nuclear safety.

5. Implementing procedures of the emergency plan, including procedures for coping with emergency conditions involving potential or actual releases of radioactivity.

6. Implementing procedures of the fire protection program.

7. Implementing procedures for the Process Control Program and Offsite Dose Calculation Manual including quality control measures.

Drills on the procedures specified in A.3 above shall be conducted as a part of the retraining program. Drills on the procedures specified in A.6 above shall be conducted at least semi-annually, including a check of communications with offsite support groups.

E. Offsite Dose Calculation Manual (ODCM)

The ODCM shall be approved by the Commission prior to initial implementation. Changes to the ODCM shall satisfy the following requirements:

1. Shall be submitted to the Commission with the Semi-Annual Radioactive Effluent release report for the period in which the change(s) were made effective. This submittal shall contain:

   a. sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information. Information submitted should consist of a package of those pages of the ODCM to be changed with each page numbered and provided with a revision date, together with appropriate analyses or evaluations justifying the change(s).

   b. a determination that the change will not reduce the accuracy or reliability of dose calculations or setpoint determinations; and

   c. documentation of the fact that the change has been reviewed and found acceptable by the Operations Committee.

2. Shall become effective upon review and acceptance by the Operations Committee.

F. Security

Procedures shall be developed to implement the requirements of the Security Plan and the Security Contingency Plan. These implementing procedures, with the exception of those non-safety related procedures governing work activities exclusively applicable to or performed by security personnel, shall be reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager. Security procedures not reviewed by the Operations Committee shall be reviewed and approved by the Superintendent, Security and Services.

G. Temporary Changes to Procedures

Temporary changes to procedures described in A, B, C, D, E and F above, which do not change the intent of the original procedures may be made with the concurrence of two individuals holding senior operator licenses. Such changes should be documented, reviewed by the Operations Committee and approved by a member of plant management designated by the Plant Manager within one month.