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Committed to Nuclear Excellence

Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

November 11, 2003

L-MT-03-085
Technical Specification 6.8.K

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555

Monticello Nuclear Generating Plant
Docket No. 50-263
License No. DPR-22

Technical Specification Bases Pages

Using the Monticello Technical Specification Bases Control Program, Monticello Technical Specification Bases pages have been changed. The affected pages are designated with the amendment applicable at the time and the suffix "a" or "b". The changes are summarized in Enclosure 1. Marked up pages applicable at the time the changes were made are provided in Enclosure 2. A final typed copy of the changed pages that are applicable, for entry into the NRC authority copy, are provided in Enclosure 3. The current copy of our list of effective pages and record of revision is attached for your information, as Enclosure 4.

Please contact John Fields at 763-295-1663 with any questions.

Thomas J. Palmisano
Site Vice President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosure 1 – Summary of Technical Specification Bases Changes (TSBC)
Enclosure 2 – Monticello Technical Specification Bases Pages Marked Up With Changes
Enclosure 3 – Revised Monticello Technical Specification Bases Pages
Enclosure 4 – Monticello Technical Specification List of Effective Pages and Record of Revision

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ENCLOSURE 1

SUMMARY OF TECHNICAL SPECIFICATION BASES CHANGE (TSBC)

Following is a summary of the bases changes forwarded herein. The changes have been processed in accordance with the Monticello Technical Specification Bases Control Program described in Technical Specification 6.8.K.

<u>TSBC-136a</u>
Technical Specification Involved – 3.7
Page affected – 182a
Summary of Change: This TSBC adds language to the Technical Specification Bases that defines the abbreviation EFCV in the bases page as Excess Flow Check Valve.

<u>TSBC-137a</u>
Technical Specification Involved – 3.6.4.D
Pages affected – 150, 151, 152, 152a, 152b
Summary of Change: This TSBC modifies the Technical Specification bases for changes made under License Amendment 137 which revised the technical specifications to be similar to alter Reactor Coolant System (RCS) leakage detection requirements and to increase operational flexibility due to the failure of RCS leakage detection equipment.

<u>TSBC-137b</u>
Technical Specification Involved – 3.13
Page affected – 225
Summary of Change: This TSBC adds language to the Technical Specification Bases that states that the "System controls on the Alternate Shutdown System (ASDS) panel" refers to portions of control circuits and instrumentation necessary solely to support ASDS functions, including equipment not located within the ASDS panel. The ASDS Limiting Condition for Operation is entered for any of the following conditions: (1) 12 Residual Heat Removal Service Water pump inoperable and (2) System controls on the ASDS panel inoperable.

ENCLOSURE 2

MONTICELLO TECHNICAL SPECIFICATION BASES PAGES MARKED UP WITH CHANGES

This attachment consists of Monticello Technical Specification bases page marked up with changes. The pages included are listed below:

6 pages follow

D. ~~Coolant Leakage~~ **Reactor Coolant System (RCS)**

1. RCS Operational Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

The low limit on increase in unidentified leakage assumes a failure mechanism of Intergranular Stress Corrosion Cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the total leakage limit. The total leakage limit considers RCS inventory makeup capability and drywell sump capacity. Drywell Equipment Drain Sump instrumentation is required to support verification of the Total Leakage limit.

With RCS unidentified or total leakage greater than the limits, actions must be taken to reduce the leak. Because the leakage limits are conservatively below the leakage that would constitute a critical crack size, 4 hours is allowed to reduce the leakage rates before the reactor must be shut down. If unidentified leakage has been identified and quantified, it may be reclassified and considered as identified leakage; however, the total leakage limit would remain unchanged.

An unidentified leakage increase of > 2 gpm within a 24 hour period is an indication of a potential flaw in the Reactor Coolant Pressure Boundary (RCPB) and must be quickly evaluated. The increase does not necessarily violate the absolute unidentified leakage limit, therefore, an option exists to allow continued reactor operation if certain susceptible components are determined not to be the source of the leakage increase within the required completion time. For an unidentified leakage increase greater than required limits, an alternative to reducing leakage increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austentic stainless steel piping that is subject to high stress or that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased leakage. This type of piping is very susceptible to IGSCC. Note also that once leakage is attributed to a specific source, that leakage can be considered to be identified and can be applied against the identified limit, rather than the unidentified limit. The 4 hour completion time is reasonable to properly reduce the unidentified leakage increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

The Surveillance Requirement (SR) associated with RCS leakage is acceptable because RCS leakage is monitored by a variety of instruments designed to provide alarms when leakage is indicated and to quantify the various types of leakage. Sump level and flow rate are typically monitored to determine actual leakage rates; however, other methods may be used to verify leakage. It is permissible to use pre-existing information, in conjunction with secondary measurements (e.g., Drywell pressure and temperature), to verify that leakage remains within limits by looking for step changes in conditions or to perform calculations to estimate leakage. The complete failure to demonstrate that RCS leakage is within limits, on the required frequency, constitutes a failure to meet this SR, notwithstanding entrance into conditions and required actions of TS 3.6.D.2.

2. RCS Leakage Detection Instrumentation

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. **The Drywell Floor Drain Sump Monitoring System instrumentation consists of one floor drain sump flow integrator, one sump level recorder and one sump fill rate computer point (rate of change). The Drywell Floor Drain Sump Monitoring System is operable when any one of these three channels is operable.** An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. **The drywell particulate radioactivity monitoring system monitors the drywell for airborne particulate radioactivity. A sudden increase in radioactivity may be attributed to RCPB steam or reactor water leakage. The drywell particulate radioactivity monitoring system is not capable of quantifying leakage rates, but is sensitive enough to indicate increased leakage rates. The drywell particulate radioactivity monitoring system provides a backup to the Drywell Floor Drain Sump Monitoring System and is capable of monitoring leakage at least as low as 10^{-9} μCi/cc radioactivity for air particulate monitoring.** Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

The Drywell Floor Drain Sump Monitoring System is required to quantify the unidentified leakage from the RCS. Thus, for the system to be considered operable, either the flow monitoring or the sump level monitoring portion of the system must be operable. Any failure of a sump monitoring subsystem should be evaluated for its impact on the ability of the associated instrumentation to measure leakage.

Since the flow integrator for each sump is not directly tied to the sump for its input signals, they are not affected in the same way as other instrumentation. However, the loss of flow through the flow integrator prevents the flow integrator from performing its intended safety function of measuring leakage, and even though its associated SRs continue to be met, it should be declared inoperable.

It should be noted that system isolation in response to Required Actions of LCO 3.7.D.2, would not render these instruments inoperable, provided the system could be unisolated as allowed by the footnote of LCO 3.7.D.2, as manual operation is allowed.

The total loss of the Drywell Floor Drain Sump Monitoring System results from the loss of all flow and level instrumentation (either directly or indirectly).

An alternate to the Drywell Floor Drain Sump Monitoring System is the drywell equipment drain sump system. Because of the physical size of the sumps, it is possible through detection or calculation to verify the required leakage limit (5 gpm) and rate limit (2 gpm/24 hrs) during the period of time it takes to actually overflow from one sump to the other. Once the drywell floor drain sump is overflowing to the drywell equipment drain sump, the drywell equipment drain sump system can be used to quantify leakage. However, the alarm settings for the equipment drain sump instruments must be reset to detect the lower limit for unidentified leakage. In this condition, all additional leakage measured by the drywell equipment drain sump system is assumed to be unidentified leakage unless the leakage has been identified and

quantified. The opposite situation is also allowed, where the equipment drain sump is allowed to overflow into the floor drain sump. In this configuration, the alarm settings need not be reset, as they would conservatively quantify all additional leakage as unidentified, unless the leakage has been identified and quantified, and alarm at the appropriate limit. The other monitoring systems provide additional indication to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. With the leakage detection systems inoperable, monitoring for leakage is degraded.

With the Drywell Floor Drain Sump Monitoring System inoperable, no other form of sampling can provide the equivalent information to quantify unidentified leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage.

With the Drywell Floor Drain Sump Monitoring System inoperable, operation may continue for 30 days. The 30 days is acceptable, based on operating experience, considering other methods of detecting leakage are available. The action requirements are modified by a footnote that allows a Mode change when the Drywell Floor Drain Sump Monitoring System is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

With the drywell particulate radioactivity monitoring system inoperable, operation may continue as long as grab samples are taken every 12 hours to analyze the drywell atmosphere. The action requirements are modified by a footnote that allows a Mode change when the drywell particulate radioactivity monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

With the required leakage detection instrumentation inoperable, no means of detecting leakage is available. This condition does not provide the required means of leakage detection. The required action is to restore one channel of the inoperable monitoring systems (Drywell Floor Drain Sump Monitoring System or drywell particulate radioactivity monitoring system) to operable status within 1 hour to regain the intended leakage detection capability. The 1-hour completion time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief valve

setpoint is established by the operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

F. Deleted

Excess Flow Check Valves (EFCVs)

With one or more penetration flow paths with one PCIV inoperable, the affected penetration must be returned to operable status or isolated within 4 hours (8 hours for MSIVs and 72 hours for ~~EFCVs~~). The 4 hour completion time is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment. The 8 hour completion time for MSIVs allows a period of time to restore the MSIVs to operable status given the fact that MSIV closure will result in a potential for plant shutdown. The 72 hour completion time for EFCVs is reasonable considering the instrument and the small diameter of the penetration piping combined with the ability of the penetration to act as an isolation boundary. With one or more penetrations with two PCIVs inoperable, either the inoperable PCIVs must be returned to operable status or the affected penetration flow path must be isolated within 1 hour.

Specification 3.7.D.3 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the control room operators to monitor the activity level of the resulting effluent by use of the Reactor Building Vent Wide Range Gas Monitors.

E. Combustible Gas Control System

The function of the Combustible Gas Control System (CGCS) is to maintain oxygen concentrations in the post-accident containment atmosphere below combustible concentrations. Oxygen may be generated in the hours following a loss of coolant accident from radiolysis of reactor coolant.

The Technical Specifications limit oxygen concentrations during operation to less than four percent by volume during operation. The maintenance of an inert atmosphere during operation precludes the build-up of a combustible mixture due to a fuel metal-water reaction. The other potential mechanism for generation of combustible mixtures is radiolysis of coolant which has been found to be small.

A special report is required to be submitted to the Commission to outline CGCS equipment failures and corrective actions to be taken if inoperability of one train exceeds thirty days. In addition, if both trains are inoperable for more than 30 days, the plant is required to shutdown until repairs can be made.

Bases 3.13:

The alternate shutdown system panel is provided to assure the capability of achieving cold shutdown, external to the control room, in the unlikely event the control room becomes uninhabitable or safe shutdown equipment in the control room or cable spreading room is damaged by fire. Control of those systems on the alternate shutdown system panel is taken when the locking master transfer switch is moved from the normal to the transfer position and each system's individual transfer switch is put in the transfer mode. When control is established at the alternate shutdown system panel no control of those systems is available from the control room and all automatic initiation signals have been disabled. The master transfer switch shall remain in the locked position at all times when not in use, being tested or being maintained. If the master transfer switch is moved to the transfer position there is an alarm in the control room.

ADD AS A NEW PARAGRAPH

"System controls on the ASDS panel" refers to portions of control circuits and instrumentation necessary solely to support ASDS functions, including equipment not located within the ASDS panel. The ASDS LCO is entered for any of the following conditions:

1. 12 RHRSW pump inoperable.
2. System controls on the ASDS panel inoperable.

ENCLOSURE 3

REVISED MONTICELLO TECHNICAL SPECIFICATION BASES PAGES

This attachment consists of the revised Monticello Technical Specification Bases pages that incorporate the change. These pages should be entered into the NRC Authority copies of Technical Specifications. The pages included are listed below:

<u>Page</u>

150
151
152
152a
152b
182a
225

7 pages follow

Bases 3.6/4.6 (Continued):

D. Reactor Coolant System (RCS)

1. RCS Operational Leakage

The allowable leakage rates of coolant from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The normally expected background leakage due to equipment design and the detection capability of the instrumentation for determining leakage was also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for unidentified leakage, the probability is small that the imperfection or crack associated with such leakage would grow rapidly. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be Pressure Boundary Leakage and they cannot be reduced within the allowed times, the reactor will be shutdown to allow further investigation and corrective action.

The low limit on increase in Unidentified Leakage assumes a failure mechanism of Intergranular Stress Corrosion Cracking (IGSCC) that produces tight cracks. This flow increase limit is capable of providing an early warning of such deterioration.

No applicable safety analysis assumes the Total Leakage limit. The Total Leakage limit considers RCS inventory makeup capability and drywell sump capacity. Drywell Equipment Drain Sump instrumentation is required to support verification of the Total Leakage limit.

With RCS Unidentified or Total Leakage greater than the limits, actions must be taken to reduce the leak. Because the leakage limits are conservatively below the leakage that would constitute a critical crack size, 4 hours is allowed to reduce the leakage rates before the reactor must be shut down. If Unidentified Leakage has been identified and quantified, it may be reclassified and considered as Identified Leakage; however, the Total Leakage limit would remain unchanged.

An Unidentified Leakage increase of >2 gpm within a 24 hour period is an indication of a potential flaw in the Reactor Coolant Pressure Boundary (RCPB) and must be quickly evaluated. The increase does not necessarily violate the absolute Unidentified Leakage limit, therefore, an option exists to allow continued reactor operation if certain susceptible components are determined not to be the source of the leakage increase within the required completion time. For an Unidentified Leakage increase greater than required limits, an alternative to reducing leakage increase to within limits (i.e., reducing the leakage rate such that the current rate is less than the "2 gpm increase in the previous 24 hours" limit; either by isolating the source or other possible methods) is to evaluate service sensitive type 304 and type 316 austenitic stainless steel piping that is subject to high stress or

that contains relatively stagnant or intermittent flow fluids and determine it is not the source of the increased leakage. This type of piping is very susceptible to IGSCC. Note also that once leakage is attributed to a specific source, that leakage can be considered to be identified and can be applied against the identified limit, rather than the unidentified limit. The 4 hour completion time is reasonable to properly reduce the Unidentified Leakage increase or verify the source before the reactor must be shut down without unduly jeopardizing plant safety.

The Surveillance Requirement (SR) associated with RCS leakage is acceptable because RCS leakage is monitored by a variety of instruments designed to provide alarms when leakage is indicated and to quantify the various types of leakage. Sump level and flow rate are typically monitored to determine actual leakage rates; however, other methods may be used to verify leakage. It is permissible to use pre-existing information, in conjunction with secondary measurements (e.g., drywell pressure and temperature), to verify that leakage remains within limits by looking for step changes in conditions or to perform calculations to estimate leakage. The complete failure to demonstrate that RCS leakage is within limits, on the required frequency, constitutes a failure to meet this SR, notwithstanding entrance into conditions and required actions of TS 3.6.D.2.

2. RCS Leakage Detection Instrumentation

Two leakage collection sumps are provided inside primary containment. Identified leakage is piped from the recirculation pump seals, valve stem leak-offs, reactor vessel flange leak-off, bulkhead and bellows drains, and vent cooler drains to the drywell equipment drain sump. All other leakage is collected in the drywell floor drain sump. Both sumps are equipped with level and flow transmitters connected to recorders in the control room. The Drywell Floor Drain Sump Monitoring System instrumentation consists of one floor drain sump flow integrator, one sump level recorder and one sump fill rate computer point (rate of change). The Drywell Floor Drain Sump Monitoring System is operable when any one of these three channels is operable. An annunciator and computer alarm are provided in the control room to alert operators when allowable leak rates are approached.

Drywell airborne particulate radioactivity is continuously monitored as well as drywell atmospheric temperature and pressure. The drywell particulate radioactivity monitoring system monitors the drywell for airborne particulate radioactivity. A sudden increase in radioactivity may be attributed to RCPB steam or reactor water leakage. The drywell particulate radioactivity monitoring system is not capable of quantifying leakage rates, but is sensitive enough to indicate increased leakage rates. The drywell particulate radioactivity monitoring system provides a backup to the Drywell Floor Drain Sump Monitoring System and is capable of monitoring leakage at least as low as 10^{-9} μCi/cc radioactivity for air particulate monitoring. Systems connected to the reactor coolant systems boundary are also monitored for leakage by the Process Liquid Radiation Monitoring System.

Bases 3.6/4.6 (Continued):

The Drywell Floor Drain Sump Monitoring System is required to quantify the unidentified leakage from the RCS. Thus, for the system to be considered operable, either the flow monitoring or the sump level monitoring portion of the system must be operable. Any failure of a sump monitoring system should be evaluated for its impact on the ability of the associated instrumentation to measure leakage.

Since the flow integrator for each sump is not directly tied to the sump for its input signals, they are not affected in the same way as other instrumentation. However, the loss of flow through the flow integrator prevents the flow integrator from performing its intended safety function of measuring leakage, and even though its associated SRs continue to be met, it should be declared inoperable.

It should be noted that system isolation in response to Required Actions of LCO 3.7.D.2, would not render these instruments inoperable, provided the system could be unisolated as allowed by the footnote of LCO 3.7.D.2, as manual operation is allowed.

The total loss of the Drywell Floor Drain Sump Monitoring System results from the loss of all flow and level instrumentation (either directly or indirectly).

An alternate to the Drywell Floor Drain Sump Monitoring System is the drywell equipment drain sump system. Because of the physical size of the sumps, it is possible through detection or calculation to verify the required leakage limit (5 gpm) and rate limit (2 gpm/24 hours) during the period of time it takes to actually overflow from one sump to the other. Once the drywell floor drain sump is overflowing to the drywell equipment drain sump, the drywell equipment drain sump system can be used to quantify leakage. However, the alarm settings for the equipment drain sump instruments must be reset to detect the lower limit for unidentified leakage. In this condition, all additional leakage measured by the drywell equipment drain sump system is assumed to be Unidentified Leakage unless the leakage has been identified and quantified. The opposite situation is also allowed, where the equipment drain sump is allowed to overflow into the floor drain sump. In this configuration, the alarm settings need not be reset, as they would conservatively quantify all additional leakage as unidentified, unless the leakage has been identified and quantified, and alarm at the appropriate limit. The other monitoring systems provide additional indication to the operators so closer examination of other detection systems will be made to determine the extent of any corrective action that may be required. The drywell particulate radioactivity monitoring system provides a backup system to the Drywell Floor Drain Sump Monitoring System. With the leakage detection systems inoperable, monitoring for leakage is degraded.

With the Drywell Floor Drain Sump Monitoring System inoperable, no other form of sampling can provide the equivalent information to quantify Unidentified Leakage. However, the drywell particulate radioactivity monitoring system will provide indication of changes in leakage.

Bases 3.6/4.6 (Continued):

With the Drywell Floor Drain Sump Monitoring System inoperable, operation may continue for 30 days. The 30 days is acceptable, based on operating experience, considering other methods of detecting leakage are available. The action requirements are modified by a footnote that allows a Mode change when the Drywell Floor Drain Sump Monitoring System is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

With the drywell particulate radioactivity monitoring system inoperable, operation may continue as long as grab samples are taken every 12 hours to analyze the drywell atmosphere. The action requirements are modified by a footnote that allows a Mode change when the drywell particulate radioactivity monitoring system is inoperable. This allowance is provided because other instrumentation is available to monitor RCS leakage.

With the required leakage detection instrumentation inoperable, no means of detecting leakage is available. This condition does not provide the required means of leakage detection. The required action is to restore one channel of the inoperable monitoring systems (Drywell Floor Drain Sump Monitoring System or drywell particulate radioactivity monitoring system) to operable status within 1 hour to regain the intended leakage detection capability. The 1 hour completion time ensures that the plant will not be operated in a degraded configuration for a lengthy time period.

The sensitivity of the sump leakage detection systems for detection of leak rate changes is better than one gpm in a one hour period. Other leakage detection methods provide warning of abnormal leakage and are not directly calibrated to provide leak rate measurements.

E. Safety/Relief Valves

The reactor coolant system safety/relief valves assure that the reactor coolant system pressure safety limit is never reached. In compliance with Section III of the ASME Boiler and Pressure Vessel Code, 1965 Edition, the safety/relief valves must be set to open at a pressure no higher than 105 percent of design pressure, with at least one safety/relief valve set to open at a pressure no greater than design pressure, and they must limit the reactor pressure to no more than 110 percent of design pressure. The safety/relief valves are sized according to the Code for a condition of MSIV closure while operating at 1775 MWt, followed by no MSIV closure scram but scram from an indirect (high flux) means. With the safety/relief valves set as specified herein, the maximum vessel pressure remains below the 1375 psig ASME Code limit. Only five of the eight valves are assumed to be operable in this analysis and the valves are assumed to open at 3% above their setpoint of 1109 psig with a 0.4 second delay. The upper limit on safety/relief

Bases 3.6/4.6 (Continued):

valve setpoint is established by the operating limit of the HPCI and RCIC systems of 1120 psig. The design capability of the HPCI and RCIC systems has been conservatively demonstrated to be acceptable at pressures 3% greater than the safety/relief valve setpoint of 1109 psig. HPCI and RCIC pressures required for system operation are limited by the Low-Low Set SRV System to well below these values.

The safety/relief valves have two functions; 1) over-pressure relief (self-actuation by high pressure), and 2) Depressurization/ Pressure Control (using air actuators to open the valves via ADS, Low-Low Set system, or manual operation).

The safety function is performed by the same safety/relief valve with self-actuated integral bellows and pilot valve causing main valve operation. Article 9, Section N-911.4(a)(4) of the ASME Pressure Vessel Code Section III Nuclear Vessels (1965 and 1968 editions) requires that these bellows be monitored for failure since this would defeat the safety function of the safety/relief valve.

Low-Low Set Logic has been provided on three non-Automatic Pressure Relief System valves. This logic is discussed in detail in the Section 3.2 Bases. This logic, through pressure sensing instrumentation, reduces the opening setpoint and increases the blowdown range of the three selected valves following a scram to eliminate the discharge line water leg clearing loads resulting from multiple valve openings.

Testing of the safety/relief valves in accordance with ANSI/ASME OM-1-1981 each refueling outage ensures that any valve deterioration is detected. An as-found tolerance value of 3% for safety/relief valve setpoints is specified in ANSI/ASME OM-1-1981. Analyses have been performed with the valves assumed to open at 3% above their setpoint of 1109 psig. The 1375 psig Code limit is not exceeded in any case. When the setpoint is being bench checked, it is prudent to disassemble one of the safety/relief valves to examine for crud buildup, bending of certain actuator members or other signs of possible deterioration.

Provision also has been made to detect failure of the bellows monitoring system. Testing of this system once per cycle provides assurance of bellows integrity.

F. Deleted

With one or more penetration flow paths with one PCIV inoperable, the affected penetration must be returned to operable status or isolated within 4 hours (8 hours for MSIVs and 72 hours for Excess Flow Check Valves (EFCVs)). The 4 hour completion time is reasonable considering the time required to isolate the penetration and the relative importance of supporting primary containment. The 8 hour completion time for MSIVs allows a period of time to restore the MSIVs to operable status given the fact that MSIV closure will result in a potential for plant shutdown. The 72 hour completion time for EFCVs is reasonable considering the instrument and the small diameter of the penetration piping combined with the ability of the penetration to act as an isolation boundary. With one or more penetrations with two PCIVs inoperable, either the inoperable PCIVs must be returned to operable status or the affected penetration flow path must be isolated within 1 hour.

Specification 3.7.D.3 requires the containment to be purged and vented through the standby gas treatment system except during inerting and deinerting operations. This provides for iodine and particulate removal from the containment atmosphere. Use of the 2-inch flow path prevents damage to the standby gas treatment system in the event of a loss of coolant accident during purging or venting. Use of the reactor building plenum and vent flow path for inerting and deinerting operations permits the control room operators to monitor the activity level of the resulting effluent by use of the Reactor Building Vent Wide Range Gas Monitors.

E. Combustible Gas Control System

The function of the Combustible Gas Control System (CGCS) is to maintain oxygen concentrations in the post-accident containment atmosphere below combustible concentrations. Oxygen may be generated in the hours following a loss of coolant accident from radiolysis of reactor coolant.

The Technical Specifications limit oxygen concentrations during operation to less than four percent by volume during operation. The maintenance of an inert atmosphere during operation precludes the build-up of a combustible mixture due to a fuel metal-water reaction. The other potential mechanism for generation of combustible mixtures is radiolysis of coolant which has been found to be small.

A special report is required to be submitted to the Commission to outline CGCS equipment failures and corrective actions to be taken if inoperability of one train exceeds thirty days. In addition, if both trains are inoperable for more than 30 days, the plant is required to shutdown until repairs can be made.

Bases 3.13:

The alternate shutdown system panel is provided to assure the capability of achieving cold shutdown, external to the control room, in the unlikely event the control room becomes uninhabitable or safe shutdown equipment in the control room or cable spreading room is damaged by fire. Control of those systems on the alternate shutdown system panel is taken when the locking master transfer switch is moved from the normal to the transfer position and each system's individual transfer switch is put in the transfer mode. When control is established at the alternate shutdown system panel no control of those systems is available from the control room and all automatic initiation signals have been disabled. The master transfer switch shall remain in the locked position at all times when not in use, being tested or being maintained. If the master transfer switch is moved to the transfer position there is an alarm in the control room.

"System controls on the ASDS panel" refers to portions of control circuits and instrumentation necessary solely to support ASDS functions, including equipment not located within the ASDS panel. The ASDS LCO is entered for any of the following conditions:

1. 12 RHRSW pump inoperable.
2. System controls on the ASDS panel inoperable.

ATTACHMENT 4

MONTICELLO TECHNICAL SPECIFICATION
LIST OF EFFECTIVE PAGES AND RECORD OF REVISION

This attachment consists of the current Monticello Technical Specification List of Effective Pages and Record of Revision. The pages included are listed below:

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A
B
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10 pages follow

MONTICELLO NUCLEAR GENERATING PLANT
APPENDIX A TECHNICAL SPECIFICATIONS RECORD OF REVISIONS

Page	Amend No.	Page	Amend No.	Page	Amend No.	Page	Amend No.
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D	115	39	129b	77	86	125	104
E	115	40	129b	78	0	126	137
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I	130a	46a	37	82	123	129	122
J	137b	47	40	82a	63	130	82
i	128	48	89	83	24	131	122
ii	137	49	128	83a	24	132	39
iii	120	50	128	84	100a	132a	122
iv	128	50a	117	85	100a	133	106
v	120	51	117	86	100a	134	133
vi	121	51a	117	87	100a	135	133
vii	122	52	128	88	100a	136	133
1	119	53	128	89	104	137	0
2	70	54	128	90	100a	138	100a
3	21	55	103	91	123	145	118a
4	102	56	102	92	100a	146	135
5	137	57	70	93	122	147	107
5a	120	58	84	94	106	148	117
6	128	58a	29	95	77	149	100a
7	128	59	128	96	77	150	137a
8	128	59a	103	97	57	151	137a
9	128	60	128	98	56	152	137a
10	128	60a	31	99	104	152a	137a
11	128	60b	62	100	100a	152b	137a
12	128	60c	30	101	122	153	100a
25a	127	60d	128	102	122	154	129a
25b	127	60e	89	103	122	155	122
25c	127	61	104	104	122	156	93
25d	127	62	117	105	122	157	130
26	5	63	117	106	79	158	132
27	81	63a	117	107	97	159	132
27a	81	64	135a	108	128	160	132
28	128	65	117	109	100a	163	130
29	128	66	119a	110	100a	164	104
30	103	67	117	111	133a	165	130
31	104	68	129b	112	130a	166	130
32	103	69	129b	113	130a	167	112
33	103	69a	129b	114	133a	168	94
34	83	70	117	115	130a	169	94
35	100a	71	100a	121	0	170	130

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171a	130	223	119
172	71	224	119
175	107	225	137b
175a	117	226	119
176	100a	229a	63
177	130	229b	104
178	100a	229c	104
179	123a	229d	63
180	130	229e	122
181	130	229u	104
182	130	229v	112
182a	136a	229v v	112
183	117	229w	112
184	100a	229ww	112
185	134	229x	112
188	104	229y	115a
189	130	229z	112
190	130	230	54
191	0	231	34
192	121	232	119
193	121	233	124
196	126a	234	119
197	121	235	115
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203	41	250	128
204	129	251	124
204a	129	252	120
205	129	253	120
206	0	254	136
207	123	255	120
208	63	256	122
209	123	257	122
209a	100a	258	134
210	100a	258a	132
211	131	259	120
212	109	260	120
213	99	261	120
216	100a	262	120
217	128		

MONTICELLO NUCLEAR GENERATING PLANT
RECORD OF TECHNICAL SPECIFICATION CHANGES AND LICENSE AMENDMENTS

NSP Page Revision (REV) No.	License DPR-22 Amend No. & Date	AEC Tech Spec Change Issuance No. and date	Major Subject
Original	-	-	Appendix A Technical Specifications incorporated in DPR-22 on 9/8/70
-	1 1/19/71	Note 1	Removed 5 MWt restriction
-	Note 2	2 1/14/72	MOGS Technical Specification changes issued by AEC but never distributed or put into effect, superseded by TS Change 12 11/15/73
1	Note 2	3 10/31/72	RHR service water pump capability change
-	Note 2	4 12/8/72	Temporary surveillance test waiver
-	2 2/20/73	Note 1	Increase in U-235 allowed in fission chambers
2	Note 2	5 3/2/73	Miscellaneous Technical Specification changes,
3	Note 2	1 4/28/71& 6 4/3/73	Respiratory Protection, & Administrative Control Changes
4	Note 2	7 5/4/73	Respiratory Protection Changes
5	Note 2	8 7/2/73	Relief Valve and CRD Scram Time Changes
6	Note 2	9 8/24/73	Fuel Densification Limits
7	Note 2	10 10/2/73	Safety Valve Setpoint Change
8	Note 2	11 11/27/73& 12 11/15/73	Offgas Holdup System, RWM, and Miscellaneous Changes
9	Note 2	13 3/30/74	8x8 Fuel Load Authorization
10	3	14 5/14/74	8x8 Full Power authorization
-	4 6/17/74	Note 1	Changed byproduct material allowance
-	6 8/20/74	Note 1	Changed byproduct material allowance
11	Note 3	Note 3 10/24/74	Inverted Tube (CRD) Limits
12	5	15 1/15/75	REMP Changes
13	7	16 2/3/75	Reactor Vessel Surveillance Program Changes
14	8	17 2/26/75	Vacuum Breaker Test Changes
15	9	18 4/10/75	Corrects Errors & Provides Clarification
-	10 7/8/75	Note 1	Increased allowed quantity of U-235
16	12	20 9/15/75	Snubber Requirements
17	11	19 9/17/75	Removed byproduct material allowance
18	13	21 10/6/75	Suppression Pool Temperature Limits
19	14	22 10/30/75	Appendix K and GETAB Limits
20	15 1/22/76 NOTE 4		Reporting Requirements
21	16 2/3/76		CRD Collet Failure Surveillance
22	17 3/16/76		NSP Organization Changes
23	NOTE 3 4/13/76		Adoption of GETAB
24	18 4/14/76		Containment Isolation Valve Testing
25	21 5/20/76		Interim Appendix B, Section 2.4 Tech. Specs.

NSP Revision (REV) No.	License DPR-22 Amend No.	& Date	Major Subject
26	19	5/27/76	Low Steamline Pressure Setpoint and MCPR Changes
27	20	6/18/76	APLHGR, LHGR, MCPR Limits
28	22	7/13/76	Correction of Errors and Environmental Reporting
29	23	9/27/76	Standby Gas Treatment System Surveillance
30	24	10/15/76	CRD Test Frequency
31	25	10/27/76	Snubber Testing Changes
32	26	4/1/77	APRS Test Method
33	27	5/24/77	MAPLHGR Clamp at Reduced Flow
34	28	6/10/77	Radiation Protection Supervisor Qualification
35	29	9/16/77	REMP Changes
36	30	9/28/77	More Restrictive MCPR
37	31	10/14/77	Inservice Inspection Changes
38	32	12/9/77	Reporting Requirements
39	33	1/25/78	Fire Protection Requirements
NOTE 1	34	4/14/78	Increase in spent fuel storage capacity
40	35	9/15/78	RPT Requirements
41	36	10/30/78	Suppression Pool Surveillance
42	37	11/6/78	8x8R Authorization, MCPR Limits & SRV Setpoints
43	NOTE 3	11/24/78	Corrected Downcomer Submergence
44	38	3/15/79	Incorporation of Physical Security Plan into License
45	39	5/15/79	Revised LPCI Flow Capability
46	40	6/5/79	Respiratory Protection Program Changes
47	41	8/29/79	Fire Protection Safety Evaluation Report
48	42	12/28/79	MAPLHGR vs. Exposure Table
49	43	2/12/80	MCPR & MAPLHGR Changes for Cycle 8 and Extended Core Burnup
50	44	2/29/80	ILRT Requirements
NOTE 1	-	8/29/80	Order for Modification of License-Environmental Qualification
NOTE 1	-	9/19/80	Revised Order for Modification of License-Environmental Qualification
51	-	10/24/80	Order for Modification of License-Environmental Qualification Records
52	-	1/9/81	Issuance of Facility Operating License (FTOL)
NOTE 1	-	1/9/81	Order for Modification of License Concerning BWR Scram Discharge Systems (License conditions removed per Amendment No. 11 dated 10/8/82)

MONTICELLO NUCLEAR GENERATING PLANT
RECORD OF TECHNICAL SPECIFICATION CHANGES AND LICENSE AMENDMENTS

NSP Revision (REV) No.	License DPR-22 Amend No.	& Date	Major Subject
NOTE 1	–	1/13/81	Order for Modification Mark I Containment
-	1	2/12/81	Revision of License Conditions Relating to Fire Protection Modifications
53	2	3/2/81	TMI Lessons Learned & Safety - Related Hydraulic Snubber Additions
54	3	3/27/81	Low voltage protection, organization and miscellaneous
NOTE 1	4	3/27/81	Incorporation of Safeguards Contingency Plan and Security Force Qualification and Training Plan into License
55	5	5/4/81	Cycle 9 - ODYN Changes, New MAPLHGR's, RPS Response time change
56	6	6/3/81	Inservice Inspection Program
57	7	6/30/81	Fire Protection Technical Specification Changes
58	8	11/5/81	Mark I Containment Modifications
59	9	12/28/81	Inservice Surveillance Requirements for Snubbers
NOTE 1	-	1/19/82	Revised Order for Modification Mark I Containment
60	10	5/20/82	Scram Discharge Volume
61	11	10/8/82	New Scram Discharge Volumes
62	12	11/30/82	RPS Power Monitor
63	13	12/6/82	Cycle 10
64	14	12/10/82	Recirc Piping and Coolant Leak Detection
65	15	12/17/82	Appendix I Technical Specifications (removed App. B)
66	16	4/18/83	Organizational Changes
67	17	4/17/83	Miscellaneous Changes
68	18	11/28/83	Steam Line Temperature Switch Setpoint
69	19	12/30/83	Radiation Protection Program
70	20	1/16/84	SRM Count Rate
71	21	1/23/84	Definition of Operability
72	22	2/2/84	Miscellaneous Technical Specification Changes
73	23	4/3/84	RPS Electrical Protection Assembly Time Delay
74	24	5/1/84	Scram Discharge Volume Vent and Drain Valves
75	25	8/15/84	Miscellaneous Technical Specification Changes
76	26	9/24/84	Cycle 11
77	27	10/31/84	RHR Intertie Line Addition
78	28	11/2/84	Hybrid I Control Rod Assembly
79	29	11/16/84	ARTS
80	30	11/16/84	Low Low Set Logic
81	31	11/27/84	Degraded Voltage Protection Logic

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NSP Revision (REV) No.	License DPR-22 Amend No. & Date		Major Subject
82	32	5/28/85	Surveillance Requirements
83	33	10/7/85	Screen Wash/Fire Pump (Partial)
84	34	10/8/85	Fuel Enrichment Limits
85	35	12/3/85	Combustible Gas Control System
86	36	12/23/85	Vacuum Breaker Cycling
87	37	1/22/86	NUREG-0737 Technical Specifications
88	38	2/12/86	Environmental Technical Specifications
89	39	3/13/86	Administrative Changes
90	40	3/18/86	Clarification of Radiation Monitor Requirements
91	41	3/24/86	250 Volt Battery
92	42	3/27/86	Jet Pump Surveillance
93	43	4/8/86	Simmer Margin Improvement
94	44	5/27/86	Cycle 12 Operation
95	45	7/1/86	Miscellaneous Changes
96	46	7/1/86	LER Reporting and Miscellaneous Changes
97	47	10/22/86	Single Loop Operation
98	48	12/1/86	Offgas System Trip
99	49	8/26/87	Rod Block Monitor
100	50	8/26/87	APRM and IRM Scram Requirements
101	51	10/16/87	2R Transformer
102	52	11/18/87	Surveillance Intervals - ILRT Schedule
103	53	11/19/87	Extension of Operating License
104	54	11/25/87	Cycle 13 and Misc Changes
105	55	11/25/87	Appendix J Testing
106	56	12/11/87	ATWS - Enriched Boron
107	57	9/23/88	Increased Boron Enrichment
108	58	12/13/88	Physical Security Plan
109	59	2/16/89	Miscellaneous Administrative Changes
110	60	2/28/89	Miscellaneous Administrative Changes
111	61	3/29/89	Fire Protection and Detection System
112	62	3/31/89	ADS Logic and S/RV Discharge Pipe Pressure
113	63	4/18/89	Miscellaneous Technical Specification Improvements
114	64	5/10/89	Containment Vent and Purge Valves
115	65	5/30/89	NUREG-0737 - Generic Letter 83-36
116	66	5/30/89	Reactor Vessel Level Instrumentation

MONTICELLO NUCLEAR GENERATING PLANT
RECORD OF TECHNICAL SPECIFICATION CHANGES AND LICENSE AMENDMENTS

NSP Revision (REV) No.	License DPR-22 Amend No. & Date		Major Subject
117	67	6/19/89	Extension of MAPLHGR. Exposure for One Fuel Type
118	68	7/14/89	SRO Requirements & Organization Chart Removal
119	69	9/12/89	Operations Committee Quorum Requirements
120	70	9/28/89	Relocation of Cycle-Specific Thermal-Hydraulic Limits
121	71	10/19/89	Deletion of Primary Containment Isolation Valve Table
122	72	11/2/89	RG 1.99, Rev 2, ISI & ILRT
123	73	5/1/90	Combined STA/LSO Position
124	74	6/5/90	Removal of WRGM Automatic ESF Actuation
125	75	10/12/90	Diesel Fuel Oil Storage
126	76	12/20/90	Miscellaneous Administrative Changes
127	77	2/15/91	Redundant and IST Testing
128	78	3/28/91	Alarming Dosimetry
125	79	4/9/91	SAFER/GESTR
130	80	8/12/91	Torus Vacuum Breaker Test Switch/EDG Fuel Oil Tank Level
131	81	4/16/92	Surveillance Test Interval Extension - Part I
132	82	7/15/92	Alternate Snubber Visual Inspection Intervals
133	83	8/18/92	Revisions to Reactor Protection System Tech Specs
134	84	1/27/93	MELLIA and Increase Core Flow
135	85	6/29/93	Revision to Diesel Fire Pump Fuel Oil Sampling Requirements
136	86	7/12/93	Revisions to Control Rod Drive Testing Requirements
137	87	4/15/94	Revised Coolant Leakage Monitoring Frequency
138	88	6/30/94	Average Planar Linear Heat Generation Rate (APLHGR) Specification & Minimum Critical Power Ratio Bases Revisions
139	89	8/25/94	Removal of Chlorine Detection Requirements and Changes to Control Room Ventilation System Requirements
140	90	9/7/94	Revisions to Radiological Effluent Specifications
141	91	9/9/94	Secondary Containment System and Standby Gas Treatment System Water Level Setpoint Change
142	92	9/15/94	Change in Safety Relief Valves Testing Requirements
143	93	7/12/95	Revised Core Spray Pump Flow
144	94	10/2/95	Standby Gas Treatment and Secondary Containment Systems
145	95	4/3/96	MSIV Combined Leakrate, and Appendix J, Option B
146	96	4/9/96	Purge and Vent Valve Seal Replacement Interval
147	97	9/17/96	Implementation of BRWOG Option I-D core Stability Solution and re-issue of pages 11, 12, 82 and 231 to reflect pages issued by NRC amendments.

MONTICELLO NUCLEAR GENERATING PLANT
RECORD OF TECHNICAL SPECIFICATION CHANGES AND LICENSE AMENDMENTS

NSP Revision (REV) No.	License DPR-22 Amend No.	Date	Major Subject
148	98	7/25/97	Bases changes on containment overpressure and number of RHR pumps required to be operable. Reissue pages 207, 209, 219, 229k, 229p, 230, 245 to reflect pages issued by NRC amendments.
149	99	10/29/97	SLMCPR for Cycle 18 and reissue pages vi, 155, 202, 207, 219, 229u
NOTE 5		11/25/97	Reissue pages a, b, g, iii, vi, 14, 25a, 155, 198y, 198z, 202, 207, 209, 219, 229k, 229p, 229r, 229u, 230, 245
150	100	4/20/98	SLMCPR for Cycle 19
NOTE 6	100a	4/30/98	Reissue all pages.
	101	08/28/98	Reactor Coolant Equivalent Radioiodine Concentration and Control Room Habitability
	102	09/16/98	Monticello Power Rerate
	103	12/23/98	Surveillance Test Interval/Allowed Outage Time Extension Program - Part 2
	104	12/24/98	Revision of Statement on Shift Length & other Misc Changes
	105	03/19/99	CST Low Level HPCI/RCIC Suction Transfer
	106	10/12/99	Revised RPV-PT Curves & remove SBLC RV setpoint
	107	11/24/99	Reactor Pressure Vessel Hydrostatic and Leakage Testing
	108	12/8/99	Testing Requirements for Control Room EFT Filters
	109	02/16/00	Safety Limit Minimum Critical Power Ratio for Cycle 20
	110	08/07/00	Transfer of Operating Authority from NSP to NMC
	111	08/18/00	Transfer of Operating License from NSP to a New Utility Operating Company
	112	08/18/00	Emergency Filtration Train Testing Exceptions and Technical Specification Revisions
	113	10/02/00	Alternate Shutdown System Operability Requirements
	114	11/30/00	Safety/Relief Valve Bellows Leak Detection System Test Frequency
	115	12/21/00	Administrative Controls and Other Miscellaneous Changes
	115a	02/13/01	Bases Change to Reflect Modification 98Q145 Installed Control Room Toxic Gas Air Supply
	116	03/01/01	Relocation of Inservice Inspection Requirements to a Licensee Program
	117	03/07/01	Reactor Water Cleanup (RWCU) System Automatic Isolation and Miscellaneous Instrumentation System Changes
	118	03/09/01	Revision of Standby Liquid Control System Surveillance Requirements
	118a	05/10/01	Bases Change - 50°F Loop Temperature, Bus Transfer & Rerate Correction

MONTICELLO NUCLEAR GENERATING PLANT
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NSP Revision (REV) No.	License DPR-22 Amend No. & Date		Major Subject
	119	04/05/01	Fire Protection Technical Specification Changes
	119a	06/28/01	Bases Change - Added information on cooldown rate
	120	07/24/01	Relocation of Radiological Effluent Technical Specifications to a Licensee-Controlled Program
	121	07/25/01	Clarify air ejector offgas activity sample point and operability requirements
	122	08/01/01	Relocation of Inservice Testing Requirements to a Licensee-Controlled Program
	122a	10/22/01	Bases Change - Remove scram setpoints sentence and correct typo
	123	10/26/01	Control Rod Drive and Core Monitoring Technical Specification Changes
	123a	10/25/01	Bases Change - Change to reflect new operation of drywell to suppression chamber vacuum breaker valve position indicating lights
	124	10/30/01	Relocation of Technical Specification Administrative Controls Related to Quality Assurance Plan
	124a	12/05/01	Bases Change - Change to reflect revised Technical Specification definition of a containment spray/cooling subsystem
	125	12/06/01	Safety Limit Minimum Critical Power Ratio for Cycle 21
	126	01/18/02	Elimination of Local Suppression Pool Temperature Limits
	126a	02/15/02	Bases Change - Change reflects relocation of sample point for the offgas radiation monitor
	127	05/31/02	Missed Surveillance Requirement Technical Specification Changes
	128	06/11/02	Changes to the Technical Specifications Revised Reference Point for Reactor Vessel Level Setpoints, Simplification of Safety Limits, and Improvement to the Bases
	128a	07/11/02	Bases Change - Correct Drywell to Suppression Chamber Vacuum Breaker Indicating Light Description
	129	08/27/02	Revise Technical Specifications and Surveillance Requirements Relating to Standby Diesel Generators
	129a	09/12/02	Bases Change - Change to Snubber Operability Description
	129b	09/12/02	Bases Change - Remove Language That Implies Trip Settings Can Be Modified By Deviation Values
	130	09/23/02	Containment Systems Technical Specification Changes
	130a	09/26/02	Bases Change - HPCI - Change Wording / HPCI & RCIC - Enhance with Wording Consistent with NUREG-1433-Rev 1

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NSP Revision (REV) No.	License DPR-22 Amend No.	Date	Major Subject
	131	10/02/02	Update the Multiplier Values for Single Loop Operation Average Planar Linear Heat Generation Rate (APLHGR)
	132	02/04/03	Conversion to Option B for Containment Leak Rate Testing
	133	02/24/03	Revision to Pressure-Temperature Curves
	133a	03/28/03	Bases Change - Adequate Reactor Steam Flow for HPCI/RCIC Testing
	134	03/31/03	One-Time Extension of Containment Integrated Leak-Rate Test Interval
	135	04/22/03	Boiling Water Reactor Vessel and Internals Project Reactor Pressure Vessel Integrated Surveillance Program
	135a	04/24/03	Bases Change - Clarify description of head cooling Group 2 valves
	136	06/17/03	Elimination of Requirements for Post Accident Sampling System
	136a	09/25/03	Bases Change - Editorial change to define the abbreviation "EFCV."
	137	08/21/03	Drywell Leakage and Sump Monitoring Detection System
	137a	10/09/03	Bases Change - RCS Leakage Requirements for TS 3.6.4.D
	137b	10/14/03	Bases Change - Clarification of system control boundary for ASDS

1. License Amendment or Order for Modification of License not affecting Technical Specifications.
2. Technical Specification change issued prior to 10 CFR revisions which require issuance of Technical Specification changes as License Amendments.
3. Modification to Bases. No Technical Specification change or License Amendment issued.
4. Technical Specification change numbers no longer assigned beginning with Amendment 15.
5. Pages reissued 11/25/97 to conform with NRC version. Revision number of effected pages not changed.
6. All pages reissued using INTERLEAF in different font. Using NRC Amendment Nos. and issue date. For Bases and Table of Contents, spelling errors corrected and editorial corrections made and all Amendment Nos. changed to 100a. For remaining Tech Spec pages, no other changes made and current Amendment Nos. used.