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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

September 16, 2005

L-MT-05-086
Technical Specification 6.8.K

U.S. Nuclear Regulatory Commission
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Monticello Nuclear Generating Plant
Docket 50-263
License No. DPR-22

Update Monticello Nuclear Generating Plant Technical Specification Bases Pages

In accordance with the provisions of Monticello Nuclear Generating Plant (MNGP) Specification 6.8.K, "Technical Specifications (TS) Bases Control Program," the Nuclear Management Company, LLC (NMC) is providing the enclosed changes for inclusion in the MNGP TS Bases.

Enclosure 1 provides a summary of the TS Bases changes. Enclosure 2 provides the current list of effective pages and records of revision (for information) and a typed copy of the revised TS Bases pages, for entry into the U.S. Nuclear Regulatory Commission Authority copy.

This letter contains no new commitments and makes no revisions to existing commitments.

John T. Conway

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Enclosures (2)

cc: Administrator, Region III, USNRC
Project Manager, Monticello, USNRC
Resident Inspector, Monticello, USNRC
Minnesota Department of Commerce

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ENCLOSURE 1

SUMMARY OF TECHNICAL SPECIFICATION BASES CHANGES

Following is a summary of the Technical Specification Bases changes (TBSC) forwarded herein. The changes have been processed in accordance with Monticello Nuclear Generating Plant (MNGP) Specification 6.8.K, "Technical Specifications (TS) Bases Control Program," and have been reviewed by the plant Operations Committee. Enclosure 2 provides a copy of the revised Bases pages.

1. **TSBC-140a**

   Technical Specification Bases Involved – 3.6/4.6 Item D.2

   Page affected – 151

   Summary of Change: The Bases for the Reactor Coolant System Leakage Detection Instrumentation incorrectly listed the drywell ventilation coolers as routed to the drywell equipment drain sump. This change corrects the Bases to reflect actual equipment configuration.

2. **TSBC-141a**

   Technical Specification Bases Involved – 3.3/4.3 Item B.3

   Page affected – 88

   Summary of Change: The Bases associated with the Rod Worth Minimizer (RWM) imply the sole reason for bypassing it is that it is physically unable to operate. In 2004, the NRC approved a report providing an improvement to the Banked Position Withdrawal Sequence acceptance criteria for control rod movement. This TBSC clarifies that the RWM may be bypassed whenever it is incapable of enforcing the control rod withdrawal / insertion sequence to be used.

3. **TSBC-141b**

   Technical Specification Bases Involved – 3.2, 3.7.E and 4.7.E

   Pages affected – 64, 182a and 190

   Summary of Changes: These changes are associated with license amendments 138 and 140. Amendment 138 removed the Combustible Gas Control System (CGCS) from the TS. During the 2005 Refueling Outage (RFO) sealing of the CGCS containment penetrations was completed. Wording pertaining to CGCS in Bases sections 3.7.E and 4.7.E may now be completely removed from the TS. Amendment 140 removed the reactor head cooling line from the TS. Bases section 3.2 was modified to clarify that the valves in this line provided containment isolation until the penetration for the reactor head cooling line was sealed during the 2005 RFO. Now that the penetration is sealed this statement can be removed.

4. **TSBC-141c**

Technical Specification Bases Involved – 3.4/4.4 Item A

Page affected – 99

Summary of Change: The Bases currently state the design objective of the Standby Liquid Control System is to provide the capability of bringing the reactor from full power to a cold shutdown xenon-free shutdown assuming that none of the withdrawn control rods can be inserted. Sufficient boron to bring the reactor from full power to a 3% delta k subcritical condition is injected (considering the other factors listed) in less than 125 minutes.

Global Nuclear Fuels (General Electric), the fuel vendor, now specifies a different requirement. The NRC approved fuel licensing report, GESTAR II states: "The Standby Liquid Control System (SLCS) is designed to provide the capability of bringing the reactor, at any time in a cycle, from a full power and minimum control rod inventory (which is defined to be at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive xenon-free state." This TBSC removes the 3% delta k subcritical condition and updates the design objective consistent with the current fuel design / licensing basis.

ENCLOSURE 2

TECHNICAL SPECIFICATION LIST OF EFFECTIVE PAGES, RECORD OF REVISION, AND BASES CHANGES

This enclosure provides numerically numbered 'change pages' for the Monticello Nuclear Generating Plant (MNGP) Technical Specification Bases page(s) incorporating the changes described herein. The affected Bases pages are designated with the amendment applicable at the time and the suffix "a."

This enclosure also provides, for information, the current, alphabetically ordered, 'change pages' for two indexes to the MNGP Technical Specifications. The first, "Appendix A, Technical Specifications Record of Revisions" provides a list of effective pages and corresponding amendment numbers. The second, the "Record of Technical Specification Changes and License Amendments," correlates between the amendment numbers and the subject of the amendment or bases changes.

The page(s) included in this enclosure and instructions for insertion into the Technical Specifications are provided below:

Remove the pages listed below and destroy.	Replace the removed pages with the pages listed below.
A	A
B	B
J	J
64	64
88	88
99	99
151	151
182a	182a
190	190

These replacement pages should be entered into the NRC Authority copy of the MNGP Technical Specifications.

9 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.
A	142	35	100a	71	100a	121	0
B	142	36	128	71a	129b	122	135
C	115	37	128	72	104	123	117
D	115	38	128	76	0	124	121
E	115	39	129b	77	86	125	104
F	115	40	129b	78	0	126	137
G	115	42	138a	79	0	126a	137
H	119	45	0	80	29	127	137
I	130a	46	70	81	3	128	42
J	142	46a	37	82	123	129	122
K	141	47	40	82a	63	130	82
i	128	48	89	83	24	131	122
ii	138	49	140	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	140	87	100a	135	133
vii	122	52	128	88	141a	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	141	95	77	149	100a
7	128	59	140	96	77	150	137a
8	128	59a	140	97	57	151	140a
9	128	60	128	98	56	152	137a
10	128	60a	31	99	141c	152a	137a
11	128	60b	62	100	141c	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	141
25d	127	62	117	105	122	157	130
26	5	63	117	106	79	158	141
27	81	63a	117	107	97	159	132
27a	81	64	141b	108	128	160	132
28	128	65	117	109	100a	163	141
29	128	66	119a	110	100a	164	141
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

MONTICELLO NUCLEAR GENERATING PLANT
APPENDIX A TECHNICAL SPECIFICATIONS RECORD OF REVISIONS

Page	Amend No.	Page	Amend No.
170	141	217	128
171	130	218	120
171a	141	223	119
172	138	224	119
175	107	225	137b
175a	117	226	119
176	100a	229a	63
177	130	229b	138
178	100a	229c	104
179	123a	229d	138
180	130	229e	122
181	130	229u	104
182	130	229v	112
182a	141b	229v v	112
183	117	229w	112
184	100a	229ww	112
185	134	229x	112
188	104	229y	115a
189	130	229z	112
190	141b	230	54
191	0	231	34
192	121	232	119
193	121	233	124
196	126a	234	119
197	121	235	115
198	121	236	115
199	51	243	128
200	129	244	124
201	129	248	59
202	129	249	142
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

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

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
	138	05/21/04	Elimination of Requirements for Hydrogen Recombiners and Hydrogen and Oxygen Monitors
	138a	06/10/04	Bases Change - Clarification of Tech Spec Table 4.1.1 Manual Scram
	139	06/02/04	Revised Analysis of Long-Term Containment Response and Net Positive Suction Head (Design Bases and USAR change)
	140	11/02/04	Revision to Technical Specification Tables 3.2.1 and 3.2.4
	140a	01/13/05	Bases Change - Removal of Drywell Vent Coolers from 3.6/4.6 Bases
	141	01/28/05	Revision to Technical Specifications Table 3.2.3 and Section 3.7/4.7
	141a	02/24/05	Bases Change - Implement Improved BPWS as Described in NEDO-33091-A
	141b	03/10/05	Bases Change - Bases Changes for License Amendments 138 and 140
	141c	03/10/05	Bases Change - Removal of 3% Delta-K from Standby Liquid Control Bases 3.4.A/4.4.A
	142	02/01/05	Deletion of Requirements for Submittal of Occupational Radiation Reports, Monthly Operating Reports, and Report of Safety/Relief Valve Failures and Challenges

Bases 3.2:

In addition to reactor protection instrumentation which initiates a reactor scram, protective instrumentation has been provided which initiates action to mitigate the consequences of accidents which are beyond the operators ability to control, or terminate a single operator error before it results in serious consequences. This set of specifications provides the limiting conditions of operation for the primary system isolation function, initiation of the emergency core cooling system, and other safety related functions. The objectives of the Specifications are (i) to assure the effectiveness of the protective instrumentation when required, and (ii) to prescribe the trip settings required to assure adequate performance. This set of Specifications also provides the limiting conditions of operation for the control rod block system.

Isolation valves are installed in those lines that penetrate the primary containment and must be isolated during a loss of coolant accident so that the radiation dose limits are not exceeded during an accident condition. Actuation of these valves is initiated by protective instrumentation shown in Table 3.2.1 which senses the conditions for which isolation is required. Such instrumentation must be available whenever primary containment integrity is required. The objective is to isolate the primary containment so that the guidelines of 10 CFR 100 are not exceeded during an accident.

The instrumentation which initiates primary system isolation is connected in a dual bus arrangement. Thus, the discussion given in the bases for Specification 3.1 is applicable here.

The low reactor water level instrumentation is set to trip when reactor water level is >7″ on the instrument. This corresponds to a lower water level inside the shroud at 100% power due to the pressure drop across the dryer/separator. This has been accounted for in the affected transient analysis. This trip initiates closure of Group 2 primary containment isolation valves. Reference Section 7.7.2.2 FSAR. The trip setting provides assurance that the valves will be closed before perforation of the clad occurs even for the maximum break in that line and therefore the setting is adequate.

The low low reactor water level instrumentation is set to trip when reactor water level is ≥-48″. This trip initiates closure of the Group 1 and Group 3 Primary containment isolation valves, Reference Section 7.7.2.2 FSAR, and also activates the ECC systems and starts the emergency diesel generators.

Bases 3.3/4.3 (Continued):

Should a control rod drop accident result in a peak fuel energy content of 280 cal/gm, less than 660 (7 x 7) fuel rods are conservatively estimated to perforate. This would result in offsite doses twice that previously reported in the FSAR, but still well below the guideline values of 10 CFR 100. For 8 x 8 fuel, less than 850 rods are conservatively estimated to perforate, which has nearly the same consequences as for the 7 x 7 fuel case because of the operating rod power differences.

The RWM provides automatic supervision to assure that out-of-sequence control rods will not be withdrawn or inserted; i.e., it limits operator deviations from planned withdrawal sequences. Reference Section 7-9 FSAR. It serves as an independent backup of the normal withdrawal procedure followed by the operator. In the event that the RWM is not capable of enforcing a particular control rod withdrawal/insertion sequence when required, it is considered to be inoperable, and a second independent operator or engineer can manually fulfill the operator-follower control rod pattern conformance function of the RWM. In this case, procedural control is exercised by verifying all control rod positions after the withdrawal of each group, prior to proceeding to the next group. Allowing substitution of a second independent operator or engineer in case of RWM inoperability recognizes the capability to adequately monitor proper rod sequencing in an alternate manner without unduly restricting plant operations. Above 10% power, there is no requirement that the RWM be operable since the control rod drop accident with out-of-sequence rods will result in a peak fuel energy content of less than 280 cal/gm. To assure high RWM availability, the RWM is required to be operating during a startup for the withdrawal of a significant number of control rods for any startup after May 1, 1974.

4. The Source Range Monitor (SRM) system performs no automatic safety system function; i.e., it has no scram function. It does provide the operator with a visual indication of neutron level. This is needed for knowledgeable and efficient reactor startup at low neutron levels. The

Bases 3.4/4.4:

A. The design objective of the standby liquid control system is to provide the capability of bringing the reactor, at any time in a fuel cycle, from full power and minimum control rod inventory (which is at the peak of the xenon transient) to a subcritical condition with the reactor in the most reactive, xenon-free state without taking credit for control rod movement. To meet this objective, the liquid control system is designed to inject a quantity of boron which produces a concentration of boron in the reactor core in less than 125 minutes sufficient to bring the reactor from full power to a subcritical condition considering the hot to cold reactivity swing, xenon poisoning and an additional 25% boron concentration margin to allow for leakage and imperfect mixing.

The time requirement (125 minutes) for insertion of the boron solution was selected to override the rate of reactivity insertion due to cooldown of the reactor following the xenon poison peak.

The ATWS Rule (10 CFR 50.62) requires the addition of a new design requirement to the generic SLC System design basis. Changes to flow rate, solution concentration or boron enrichment to meet the ATWS Rule do not invalidate the original system design basis. Paragraph (c)(4) of 10 CFR 50.62 states that:

> "Each boiling water reactor must have a Standby Liquid Control System (SLCS) with a minimum flow capacity and boron content equivalent in control capacity to 86 gallons per minute of 13 weight percent sodium pentaborate solution" (natural boron enrichment).

The described minimum system parameters (equivalent to 24 gpm, 10.7% concentration and 55 atom percent Boron-10 enrichment) will ensure an equivalent injection capability that meets the ATWS rule requirement.

Boron enrichment concentration, solution temperature, and volume (including check of tank heater and pipe heat tracing system) are checked on a frequency to assure a high reliability of operation of the system should it ever be required. Only one of the two standby liquid control pumping circuits is needed for proper operation of the system. If one pumping circuit is found to be inoperable, there is no immediate threat to shutdown capability, and reactor operation may continue while repairs are being made. A reliability analysis indicates that the plant can be operated safely in this manner for ten days. For additional margin, the allowable out of service time has been reduced to seven days.

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, and bulkhead and bellows 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.

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. (Deleted)

Bases 4.7 (Continued):

will be in the isolation position should an event occur. This required action does not require any testing or device manipulation. Rather, it involves verification that those devices outside containment and capable of potentially being mispositioned are in the correct position. The completion time of "monthly" for devices outside containment is appropriate because the devices are operated under administrative controls and the probability of their misalignment is low. For the devices inside primary containment, the time period specified "prior to entering Startup or Hot Shutdown from Cold Shutdown, if primary containment was deinerted while in Cold Shutdown, if not performed in the previous 92 days" is based on engineering judgement and is considered reasonable in view of the inaccessibility of the devices and other administrative controls ensuring that device misalignment is an unlikely possibility.

The surveillance requirements are modified by a footnote allowing both active and passive isolation devices, used to isolate a penetration, that are located in high radiation areas can be verified by use of administrative means. Allowing verification by administrative means is considered acceptable, since access to these areas is typically restricted. Therefore, the probability of misalignment of these devices, once they have been verified in the proper position, is low.

The containment is penetrated by a large number of small diameter instrument lines. A program for the periodic testing (see Specification 4.7.D) and examination of the valves in these lines has been developed and a report covering this program was submitted to the AEC on July 27, 1973.

The main steam line isolation valves are functionally tested on a more frequent interval to establish a high degree of reliability.

E. (Deleted)