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Monticello Nuclear Generating Plant
Operated by Nuclear Management Company, LLC

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

Response to Request for Additional Information Related to License Amendment Request for Full Scope Alternative Source Term

References: 1) NMC letter to NRC, "License Amendment Request: Full Scope Application of an Alternative Source Term," (ADAMS Accession No. ML052640366) dated September 15, 2005.

2) NMC letter to NRC, "Full Scope Alternate Source Term-Supplemental Information," (ADAMS Accession No. ML 061310445) dated April 13, 2006.

During the review of the Monticello application for a Full Scope Alternative Source Term (AST), the Nuclear Regulatory Commission (NRC) provided Nuclear Management Company, LLC (NMC) with questions regarding the application. The questions were transmitted to NMC via e-mails dated June 29, July 13 and August 3, 2006, and were clarified during conference calls held on July 11, July 17 and August 4, 2006.

NMC responses to the questions are included in Enclosures 1, 2 and 3. Enclosure 1 contains the responses to questions regarding the accident dose assessment review. Enclosure 2 contains the responses to questions regarding the meteorology review and Enclosure 3 contains the responses to questions regarding the containment ventilation systems review. As part of this submittal, Enclosure 5 includes reference drawings and one calculation which were previously provided to NRC staff to assist in the technical review.

In addition, MNGP has determined that the sector used for the Control Room elevated release X/Q calculation in CA-04-037 Rev 2 (MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis) was incorrect. The sector used was Southeast (SE);

A001

the opposite Northwest (NW) sector should have been used. The details and evaluation of this minor error are provided as Enclosure 4.

Summary of Commitments

This letter contains no new or revised commitments.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on August 22, 2006.

John T. Conway

John T. Conway
Site Vice-President, Monticello Nuclear Generating Plant
Nuclear Management Company, LLC

Enclosures (5)

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

ENCLOSURE 1

NMC Response to RAI Questions (Accident Dose Assessment)

NRC Question 1a:

What is the basis for the assumed reduction in leakage through the main steam isolation valves (MSIVs) at 24 hours and 72 hours in the loss of coolant accident (LOCA) dose analysis?

NMC Response 1a:

The basis for the reduction in the assumed MSIV leakage rate at 24 hours and 72 hours is plant-specific analysis of the post-LOCA primary containment pressure/temperature profile.

NMC Calculation CA-03-099 Rev 1, "Drywell Temperature and Pressure EQ Profiles", provides drywell pressure and temperature response for EQ purposes. The calculation reviewed data from a variety of scenarios from existing MNGP-specific containment response analyses and determined conservative bounding drywell pressure and temperature values for EQ.

These bounding values were compared with the latest evaluation of containment response (GE-NE-0000-00002-8817-01 Rev. 2, August 2003, documented in USAR Section 5.2), including scaling the GE-NE response from 1775 to 1880 MWt, and determined to be still bounding. Therefore the pressure and temperature values from CA-03-099 were considered to provide a conservative containment response basis for reduction in MSIV and primary containment leakage at 24 and 72 hours, with the exception that the drywell pressure for 0-1 days is assumed to be the design accident pressure of 42 psig, consistent with guidance in RG 1.183 Appendix A Section 3.7 (the drywell peak pressure evaluated in CA-03-099 does not reach this pressure).

Using these post-LOCA pressure/temperature conditions and the critical mass flux (lbm/sec-ft^2) Equation No. 2.60 from Frederick J. Moody, "Introduction to Unsteady Thermofluid Mechanics", the ratio of mass fluxes is calculated for the MNGP pressures and temperatures at times greater than 24 hours post-LOCA for both steam and air assuming an ideal gas (gas properties taken from Table B.6 of Sonntag & Van Wylen, "Introduction to Thermodynamics: Classical and Statistical").

A check of the critical pressure is also performed to determine if DW pressure is less than the critical pressure needed for sonic (critical) flow (Frederick J. Moody, "Introduction to Unsteady Thermofluid Mechanics" Eqn. 2.59). Leakage rates for DW pressures below the critical pressure are not calculated, but would be less than the critical mass flux calculated above.

Results of these critical mass flux ratio calculations, based on the MNGP analyzed LOCA pressure/temperature profile, indicate that the leakage flow rate at 24 hours is reduced to 61% of the design leakage value and is reduced further at 72 hours to 50% of the design leakage value.

[Reference CA-04-038, Rev 0, "MNGP AST – LOCA Radiological Consequence Analysis": Assumption 3.9 (pg A-22) and Attachment 102, "Pressure Decay - Reduced Leakage Rate.xls". CA-04-038 Rev 0 and attachments were provided in NMC AST Full-Scope Submittal dated September 15, 2005.]

NRC Question 1b:

The submittal indicates that the reduction in leakage into the main steam lines is consistent with the reduction in primary containment leakage based on the reduction of primary containment pressure. How does the reduction in primary containment pressure lead to a reduction in MSIV leakage?

NMC Response 1b:

For the design basis LOCA (double-ended large pipe break) following the blowdown of the initial RPV/RPS inventory, the long term reactor vessel pressure profile is assumed to be essentially the same as that of the primary containment. Thus, the driving pressure for leakage (Primary, MSIV, Secondary Containment Bypass), is the post LOCA pressure within the primary containment.

NRC Question 2:

Does the LOCA analysis modeling of iodine aerosol deposition in the MSIV leakage pathway account for the preceding removal of aerosols in the primary containment by natural deposition? Discuss MNGP's interpretation of expected aerosol size distribution behavior over time vs. that assumed by the AEB-98-03 model.

NMC Response 2:

Yes, the iodine aerosol deposition modeling in the MSIV leakage pathway does account for preceding removal of aerosols in primary containment by natural deposition.

The RADTRAD model used for the assessment of LOCA dose via the MSIV leakage pathway assumes 10% Powers – BWR natural deposition correlation within the primary containment volume. MSIV leakage is taken directly from the described primary containment volume.

For the AST LOCA, all activity release from the core is assumed to be released to the primary containment within the first 2-hours post LOCA. Therefore, using the RADTRAD primary containment volume with Powers – BWR natural deposition as the source of MSIV leakage is both justifiable and reasonable.

The model based on AEB-98-03 guidance is intended to be conservative in estimating total deposition based on a deposition velocity statistically determined from drywell expected distributions of aerosol density, diameter, and shape factors. Since the MNGP Main Steam Line (MSL) aerosol deposition model is a lumped-single volume, there is no aerosol size distribution behavior impact on the model itself as the deposition volume's outlet flow empties directly into the condenser and not into other pipe deposition volumes with deposition rates based on the first volume's exit conditions. Thus, this single-volume model does not permit simulation of deposition induced changes to the initial aerosol size distribution as a function of time for a unit volume of aerosol that transits through the model. Simulation of the physical changes to the aerosol constituents would require a multi-volume model to account for time-dependent changes to control volume characteristics (i.e., loss of mass, redistribution of remaining particles, etc.). However, it is expected that a mechanistic multi-compartment pipe deposition model would yield higher deposition rates as the downstream volumes further remove additional aerosol particles not deposited in upstream volumes. The model, therefore, does not account for the effect of time-dependent aerosol mass depletion through the MSL drain system, the effect of pipe bends, condensation, or other non-uniformities that could stimulate additional droplet growth and enhance aerosol deposition. Ultimately, a realistic expectation might be for a more dilute aerosol consisting of only the smallest particles entering the condenser volume wherein additional depletion effects act on the remaining aerosol.

Thus, the dose consequence predicted from the lumped-single volume MNGP MSL iodine aerosol pipe deposition model is expected to be greater than that expected from a multi-compartment aerosol depletion model that takes into account all or most of the aerosol depletion mechanisms.

[Reference CA-04-038 Rev 0, Section 6.1 (RADTRAD models) and Figure 11]

NRC Question 3a:

Was potential iodine revolatilization accounted for in the modeling of iodine removal for the LOCA MSIV leakage pathway?

NMC Response 3a:

No. Potential revolatilization of iodines was not explicitly incorporated into the LOCA MSIV leakage pathway model. Previous evaluations have shown the revolatilization mechanism to be of minor significance in LOCA dose analysis.

The temperature dependent elemental iodine resuspension rates as reported in J. E. Cline's "MSIV Leakage Iodine Transport Analysis" are orders of magnitude lower than the particulate deposition rates calculated using the methodology in AEB 98-03. Considering the conservative nature of the pipe temperature calculation methodology used (see response to Question 3b), the potential non-conservatism of not incorporating iodine revolatilization is deemed more than compensated.

NRC Question 3b:

Does the piping temperature profile used to determine the aerosol deposition efficiency include heating from aerosols deposited in the piping?

NMC Response 3b:

No. The piping temperature profile analysis performed to support the temperature values used in the LOCA MSIV pathway dose analysis does not explicitly model decay energy from deposited aerosols. However, the analysis performed to generate the temperature profile used to evaluate LOCA dose is very conservative. Boundary conditions were explicitly chosen to maximize heat addition to the piping pathway while heat transfer boundary conditions and heat transfer coefficients were chosen to minimize heat transfer from the piping pathway. Thus, modeled pipe temperatures used in the LOCA MSIV pathway dose analysis are much higher than would be predicted based on a more realistic set of boundary conditions and would more than offset the small contribution to the pipe wall temperature resulting from the decay heat of the deposited material.

Additionally, there are significant conservatisms in the MSL thermal profile model for MNGP:

- The drywell/inside pipe temperature and pressure conditions used are the conservative values determined in MNGP Calculation CA-03-099 discussed in Response 1a above. Room temperatures for the Main Steam Tunnel are assumed at values which bound the room temperatures expected during heat-up post-LOCA.

- The Main Steam Tunnel temperatures are used to model heat transfer to the environment for the entire pipe, with MST room temperatures ranging from 175°F to 126°F. A significant portion of the MSL piping is located in the large condenser bay room. This room is more open and cooler, with an expected high temperature of 130°F.

- The MSL insulation has many penetrations that connect directly to the piping but are not credited as heat losses for calculating the thermal profile. These include branch piping, valve yokes and un-insulated structural pipe supports that act to dissipate heat directly to the surrounding atmosphere. For example, an engineering-science reference table gives a combined natural convection and radiation heat transfer coefficient of 2.25 BTU/hr-ft^2-°F for 100°F mean temperature differential (Ts – Ta) of a 3-inch diameter horizontal cylinder (a size comparable to heavy pipe support tube steel). At a 100°F effective temperature difference, a 1.2 ft^2 surface dissipates 273 BTU/hr, or about 18 inches of cylinder length. This is only a small fraction of the total support structures and attachments combined surface area. Thus, the heat gain to the piping from deposited iodine decay is greatly exceeded by the heat dissipated through the MSL piping attachments.

- The temperature profile was held constant for the first 12 hours post-LOCA although the model predicts significant temperature decrease during that period. The temperature profile was held constant after 20 days although the model predicts continuing decrease after that time.

When questioned by NRC about effects of heating from deposited iodine species, during review of a submittal made for the Clinton Nuclear Station, Clinton responded that the calculated decay heat value from deposited iodine species is approximately equal to an 80-watt incandescent light bulb, or about 273 BTU/hr. The basis for this value is a 100 scfh leakage flow rate, 100 percent of the iodine species in the leakage flow assumed to be deposited, and the iodine activity of the Clinton source term in curies converted to watts. By comparison, Clinton Nuclear Station has a license rated thermal power (RTP) of 3,473 megawatts and the MNGP LOCA source term assumes a thermal power of 1,880 megawatts. Since the source term is directly proportional to core power, the estimated decay heat for the same leak rate in the MNGP model would be about that of a 43-watt bulb or 147 BTU/hr.

Based on the above discussion and the conservatism in the piping temperature profile used, the effect of any decay heat on pipe inner surface temperature due to iodine decay energy absorption is insignificant to the evaluation of iodine deposition, and also to any re-suspension/re-evolution of deposited iodine.

The GOTHIC model, methodology, and assumptions used in the development of the MSIV pathway piping temperature profile are essentially the same as that used in the Brunswick Steam Electric Plant (BSEP) TS amendment approved on March 6, 2006 (Amendment 239 and 267).

[Reference CA-04-038, Rev 0, Assumption 3.10; CA-05-134, Rev 0, "Post-LOCA Steam Pipe Internal Temperature 30-Day Profile for Radiological Dose Analysis" (incorporates AAC Calculation MNGP-012 Rev 0)].

NRC Question No. 4:

Please provide the control room shine dose calculations, including any diagrams that indicate the location of the source(s) and receptor(s).

<u>NMC Response No. 4:</u>

MNGP Calculation CA-05-130, Rev 0, "Post LOCA Direct Dose to the Control Room from External Sources", is provided in Enclosure 5. A series of annotated plant drawings is also included as an overall view of the relative positions of the sources and receptors as well as the plant position column numbering referenced in the diagrams in CA-05-130. Supplied drawings: NF-36056-1, Rev H; NF-36057, Rev L; NF-36063, Rev A. (See Enclosure 5)

ENCLOSURE 2
NMC Response to RAI Questions (Meteorology)

NRC Question No. 1:

Section 4, Atmospheric Dispersion, of the March 7, 2006 NRC Regulatory Issue Summary 2006-04 (RIS 2006-04), Experience with Implementation of Alternative Source Terms, states that valid wind direction data in the ARCON96 format should range from 1° to 360°. Fields of "nines" (e.g., 999) should be used to indicate invalid or missing data. The staff notes that the Monticello hourly meteorological data for the 1998-2002 period generally contains fields of nines to indicate invalid data, but also has some zeros in the wind direction and wind speed fields. Are these data (i.e., the zeros) valid data? RIS 2006-04 further states that the jfds used as input to the PAVAN computer code should have a large number of wind speed categories at the lower wind speeds in order to produce the best results (e.g., Section 4.6 of NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," suggests wind speed categories of calm, 0.5, 0.75,1.0, 1.25,1.5, 2.0, 3.0, 4.0, 5.0, 6.0, 8.0 and 10.0 meters per second). What is the wind speed and designation for calm (e.g., less than 0.5 meters per second based upon instrument thresholds)? The staff is attempting to generate a jfd using these categories and is attempting to input valid data only.

NMC Response No. 1:

For the 43 Meter ARCON96-formatted data provided in the AST Full Scope submittal dated September 15, 2005 (provided as attachments in MNGP Calculation CA-04-037, Rev 2, "MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis"), not all invalid data was formatted as "99"s since "99"s were entered only for missing data. Therefore, for these data files, wind directions given as "0" should be considered as invalid data per RIS 2006-04. Wind speeds given as "0" may be calm or may be invalid data. Two periods were identified as containing invalid data due to instrument maintenance: June 14-21, 2000 and August 5-8, 2001. All data for all levels for these periods should be considered as invalid.

The 100 Meter ARCON96-formatted data supplied as part of the AST Full Scope Supplemental submittal was formatted with "99"s in all fields where the stability class for that hour was invalid or "0". The periods of invalid data in June 2000 and August 2001 discussed above were formatted as all "99"s.

For wind speed categories, MNGP followed the guidance provided in RG 1.183 Section 5.3 and RG 1.145 Section 1.1, which refer to RG 1.23. RG 1.23 details the binning categories used by MNGP for jfd tables, which are the same categories given in RG 1.21.

The joint frequency distribution tables used as input to PAVAN were generated by the MNGP MIDAS computer code. MIDAS defines calm as 0.3 meters per second or less (less than 0.4 meters per second). This value is consistent with the instrument thresholds over the five-year period of data collection (0.3 m/s through June 2000; 0.1 m/s after).

NRC Question No. 2:

What wind speed values were used as an input to Midas to define the categories in the Monticello wind speed, wind direction and atmospheric stability joint frequency distributions (jfds) for both the Regulatory Guide 1.21 and non-Regulatory Guide 1.21 tables? For example, if two consecutive categories are defined as 4-7 miles per hour (mph) and 8-12 mph, what is the value defining the lower category (e.g., 7 mph, 7.5 mph, 8 mph)?

NMC Response No. 2:

The following wind speed categories were used for the joint frequency distribution tables:

RG 1.21:
Calm, >Calm – 3.5, 3.6 – 7.5, 7.6 – 12.5, 12.6 – 18.5, 18.6 – 24.5, 24.6 – up
Non-RG 1.21:
Calm, >Calm – 3.5, 3.6 – 7.5, 7.6 – 12.5, 12.6 – 18.5, 18.6 – 24.5. 24.6 – 32.5, 32.6 - up

The value defining the lower category for the above example would be 7.5 mph.

NRC Question No. 3:

Please provide the calculations for the atmospheric dispersion factors (χ/Q values) used in the Monticello alternative source term license amendment request for new χ/Q values. Also, provide a site plan generally drawn to scale and showing true North, with postulated release/receptor pairs highlighted.

NMC Response No. 3:

MNGP calculations CA-04-036, Rev 1, "MNGP AST - Offsite Post-Accident Atmospheric Dispersion Analysis," and CA-04-037, Rev 2, "MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis," determined the atmospheric dispersion factors used in the AST Full Scope LAR dated September 15, 2005. These calculations were provided in their entirety on Disk 2 of the LAR submittal.

A site plan is included in Enclosure 5. The plan is drawn to scale, indicates true North, and shows the release points and receptors assumed in the MNGP radiological analyses.

NRC Question No. 4:

Which χ/Q values were used to model unfiltered inleakage into the control room?

NMC Response No. 4:

X/Q values were chosen to provide the most conservative source for Control Room unfiltered inleakage, Control Room normal ventilation, and Control Room emergency ventilation. For a given release, the limiting X/Q was determined and then used to model all three pathways.

The following pathways to the Control Room were identified by system evaluation:

a. The Control Room intake is used for both normal Control Room ventilation unfiltered intake and Emergency Control Room ventilation filtered intake;

b. The Control Room intake was identified as a likely source of inleakage due to ductwork carrying unfiltered air passing through the Control Room envelope;

c. The Administration Building intake was identified as a source of inleakage since it supplies unfiltered air to rooms adjacent to the Control Room.

Control Room tracer gas habitability testing was performed in June 2004 and confirmed the inleakage sources identified above.

For each release point, dispersion factors were calculated for both the Control Room Intake and the Administration Building intake. The limiting dispersion factor was then chosen for each release point, and was used to model all pathways to the Control Room from that release point (CR normal unfiltered ventilation, emergency filtered ventilation, and unfiltered inleakage).

Reference the individual DBA calculation design inputs which were provided on Disk 2 of the MNGP AST Full Scope LAR submittal (CA-04-038, Rev 0, "MNGP AST - LOCA Radiological Consequence Analysis"; CA-04-039, Rev 0, "MNGP AST - MSLBA Radiological Consequence Analysis"; CA-04-040, Rev 0, "MNGP AST - CRDA Radiological Consequence Analysis"; and CA-04-041, Rev 1, "MNGP AST - FHA Radiological Consequence Analysis").

NRC Question No. 5:

Do the accident scenarios and generated χ/Q values model the limiting doses considering multiple release scenarios (e.g., including those due loss of offsite power or other single failure)?

NMC Response No. 5:

Yes, multiple release scenarios were considered. The accident doses reported in Enclosure 3 of the NMC AST Full-Scope License Amendment Request dated 9/15/05 include the contributions from the most limiting scenarios.

The LOCA analysis assumes loss of offsite power and single failures as directed by RG 1.183 Section 5.1.2. A positive pressure period is modeled which assumes both LOOP and single failure of a Standby Gas train, during which any releases to the Secondary Containment are then released directly to the environment as a ground release (dispersion coefficient used is RB Nearest Wall). Use of the positive pressure period release followed by an elevated stack release via SBGT is the limiting scenario for releases from the Secondary Containment for the LOCA.

The CRDA analysis includes a release via the steam jet air ejectors directly to the stack with no credit for holdup in the offgas storage system. A failure of radiation monitors to isolate the SJAEs is assumed since the isolation is non-safety-related. The CRDA analysis also considers a release from the isolated condenser. The SJAE release is the limiting scenario for the CRDA.

The FHA analysis assumes a ground-level release from the Reactor Building Vent. In the February 28, 2005 MNGP response to NRC RAIs, multiple release scenarios for the FHA were discussed. This discussion remains relevant for the FHA analysis submitted with the MNGP AST Full Scope LAR. The Reactor Building Vent release is the limiting scenario for the FHA.

NRC Question No. 6:

When assessing meteorological factors that may affect secondary containment drawdown capability, is high wind speed, only, the limiting case (e.g., temperature is not a factor)?

NMC Response No. 6:

The following meteorological factors were addressed in the secondary containment drawdown (or positive pressure period, PPP) calculation: wind speed, relative humidity, and ambient temperature effects.

a. Section 4.3 of Appendix A of Regulatory Guide 1.183 requires consideration of high winds on the secondary containment ability to maintain negative pressure. Specifically, RG 1.183 states that the wind speed to be assumed is "the 1-hour average value that is exceeded only 5% of the total number of hours in the data set." At the 43 meter elevation (1072 ft., representative of the RB), meteorological tower data from 1998 through 2002 used in the radiological consequences analyses indicates that the wind speed exceeds 24 mph only 1.4% of the total sample. Section 5.3.5 of the MNGP USAR indicates that at wind speeds up to 35 mph there is little ex-filtration from the reactor building at –0.25 in WG. Further, the MNGP secondary containment surveillance tests (used to confirm the design inleakage rate used in the drawdown calculation) correct the measured RB pressure for wind conditions. Therefore, no further consideration was given to high wind effects.

b. Atmospheric relative humidity was assumed at 0%. Sensitivity studies demonstrate that this is conservative for PPP calculations.

c. The calculation addresses temperature effects as noted in NRC Information Notice No. 88-76, "Recent Discovery of a Phenomenon Not Previously Considered in the Design of Secondary Containment Pressure Control". Site ambient minimum temperature for both summer and winter operations were based on temperatures exceeded 5% of the time during the AST implementation meteorological data for the years of 1998-2002, in accordance with RG 1.183, Appendix A Section 4.3. The summer and winter low temperatures were 50.9°F and -3.5°F respectively. Inleakage air temperature and atmospheric pressure conditions (as affected by air density) were adjusted based on these temperatures. Experience in previous PPP calculations indicates that using the lowest boundary condition temperature results in a slightly longer PPP due to air density differences. For the purpose of this calculation, both high and low temperature limits were checked and the lowest temperature limit confirmed to result in a longer design PPP. The PPP calculations were performed at bounding summer and winter conditions and the worst case selected.

Based on the above discussion, ambient temperature was determined to be the limiting factor in the Secondary Containment drawdown assessment.

ENCLOSURE 3
NMC Response to RAI Questions (Containment Ventilation)

NRC Question No. 1:

Question withdrawn by NRC staff.

NMC Response No. 1:

No response required.

NRC Question No. 2:

Question withdrawn by NRC staff.

NMC Response No. 2:

No response required.

NRC Question No. 3:

Please state if the whole system is subject to Inservice Inspection requirements of ASME Boiler and Pressure Vessel Code Section 11 or other plant specific Inservice Inspection program that provides for periodic testing and examination of the condition of the system. Information on the testing of pumps, valves and frequency of test would be useful.

NMC Response No. 3:

The Standby Liquid Control (SLC) system is included in the MNGP Section XI IST program (ADAMS Accession Numbers ML023370107 and ML023370010). As specified in Attachments 7 and 8 of the program plan, tests and testing frequencies are as follows:

- *Pumps*: flow test quarterly; flow, discharge pressure and vibration tests every 2 years.
- *Relief Valves*: tested once per 10 years minimum (group sampling every 48 months).
- *Explosive (squib) valves*: tested once per 10 years minimum (group sampling every 48 months) per IST program; TS requires pathway testing every 24 months (staggered basis) which includes exploding squib.
- *Discharge check valves*: disassembled and inspected once per 8 years minimum, group sampling every RF outage.
- *Containment isolation check valves*: leakage test per App J program (maximum interval of 60 months); tested open and tested closed every RF outage.
- *Injection line manual isolation valve*: position indication test every two years.

NMC has implemented a Risk-Informed In-Service Inspection (RI-ISI) program at MNGP (ADAMS Accession Number ML0202403810). Table 3.1 of the RI-ISI Program identifies the SLC system as within the scope of the program. The actual segments or elements subject to examination for RI-ISI are selected and inspected consistent with the requirements outlined in the program.

NRC Question No. 4

Please clarify the transport of sodium pentaborate from the reactor vessel to the suppression pool after SLC injection. The staff understands that once reactor level is restored after a LOCA, the LPSI pump(s) are switched to suppression pool recirculation and cooling and that reactor vessel level is maintained by the core spray system. This would occur at about 10 minutes after the LOCA. SLC injection may not occur until sometime later (maybe two hours) when the presence of fuel damage is detected. Is the flow supplied by core spray systems sufficient to sweep the sodium pentaborate from the reactor vessel to the suppression pool in a timely fashion? The path of the flow should be considered to assure that it sweeps the whole vessel. The staff is concerned that much of the sodium pentaborate would remain in the reactor vessel and not be available to buffer the suppression pool pH. Other plants have opted to revise procedures to run the LPSI pumps in the injection mode after SLC injection to assure transport to the suppression pool. Would a revision to Monticello's procedures be reasonable?

NMC Response No. 4:

Sodium pentaborate is injected into the reactor vessel through a vertical sparger which is located in the bottom head region, mounted on the inside of the vessel wall and rising up to an elevation about one foot below the core plate. The sparger section has 8 outlet holes 1/4 inch in diameter spaced 6 inches apart and is capped at the top. The sparger serves to aid in initial dispersal of the pentaborate by providing improved distribution. An annotated drawing showing the location of the sparger in the reactor vessel is included in Enclosure 5 (NX-7831-197-1, Rev D, "Reactor Vessel and Internals").

SBLC injection will begin to enter the suppression pool within one hour and all injection will be complete within 2 hours post-accident.

NMC performed a quantitative assessment of the turnover rate within the reactor vessel with respect to sodium pentaborate injection. The volume assumed was the total reactor vessel internal volume up to 2/3 core height minus the volume of the annulus region. Flow assumed was one core spray pump injecting at 2,700 gpm minus the core boiloff rate of 300 gpm for a total of 2,400 gpm.

The assessment demonstrated that the fluid in the reactor vessel turns over at least five times per hour, thus assuring adequate mixing and transport of the sodium pentaborate.

NRC Question No. 5

It is stated in the submittal that the Maintenance rule applies to the isolation check valves and the initiation switch (the two items that were identified as not single failure proof). Does the Maintenance rule apply to other components in the system such as pumps, tanks, heat tracing, relief valves and instrumentation?

NMC Response No. 5:

The SLC system is included in the MNGP Maintenance Rule (MR) Program scope. The SLC system availability and functional failures are tracked. The primary function of the SLC System, as described by the Maintenance Rule System Basis Document, is to bring the reactor to a shutdown condition at any time in the reactor core life even if any or all withdrawn control rods are unavailable for insertion. (Note that this basis will be updated to reflect the AST suppression pool pH control function of SLC when appropriate.)

The MR Program tracks function availability, as opposed to specific component availability. Components necessary to accomplish the specified system function are identified in the site equipment database as Maintenance Rule related. For the SLC system, this includes pumps, pressure accumulators, discharge check valves, squibs, discharge valve to vessel, heat tracing, tank heater, relief valves, containment isolation check valves, and initiation switch. Verification of inclusion in the Maintenance Rule Program for the aforementioned components was performed using the MNGP equipment database. Indication instrumentation and tanks are not within the program scope. All the SLC components in the flow path that are necessary to support injection of sodium pentaborate are within the program scope.

ENCLOSURE 4
X/Q Recalculation Description

NMC has determined that the sector used for the Monticello Control Room elevated release X/Q calculation in CA-04-037 Rev 2 (MNGP AST - CR/TSC Post-Accident Atmospheric Dispersion Analysis) was incorrect. The sector used was Southeast (SE); the opposite Northwest (NW) sector should have been used.

The X/Q was recalculated using the correct sector. Results are given below:

ELEVATED RELEASE χ/Q (sec/m^3) PAVAN-PC CR Intake Location			
Time Period	**χ/Q SE**	**χ/Q NW**	**Difference**
Fumigation	3.37E-04	3.37E-04	No change
0-2 hours	3.73E-06	3.77E-06	+1.1%
0-8 hours	5.62E-07	5.74E-07	+2.1%
8-24 hours	2.20E-07	2.24E-07	+1.8%
1-4 days	2.88E-08	2.90E-08	+0.7%
4-30 days	1.56E-09	1.54E-09	-1.3%

ELEVATED RELEASE χ/Q (sec/m^3) PAVAN-PC Admin Bldg. Intake Location			
Time Period	**χ/Q SE**	**χ/Q NW**	**Difference**
Fumigation	3.59E-04	3.59E-04	No change
0-2 hours	4.02E-06	4.06E-06	+1.0%
0-8 hours	5.63E-07	5.75E-07	+2.1%
8-24 hours	2.13E-07	2.17E-07	+1.9%
1-4 days	2.58E-08	2.60E-08	+0.8%
4-30 days	1.25E-09	1.24E-09	-0.8%

This error does not affect the Control Room ground-level X/Qs calculated in CA-04-037 Rev 2. The EAB/LPZ X/Qs calculated in CA-04-036 Rev 1 (MNGP AST - Offsite Post-Accident Atmospheric Dispersion Analysis) are also unaffected since the proper sectors and sector distances were used in that calculation.

This error occurred because the calculation preparer used the ARCON96 directional convention for input to both the ARCON96 and PAVAN programs when calculating Control Room X/Qs. ARCON96 specifies input as the direction from the receptor to the source, while PAVAN specifies input of the direction (as a sector) from the source to the receptor.

ENCLOSURE 4
X/Q Recalculation Description

The CR elevated X/Q was used as input for dose assessments of the LOCA and CRDA accidents. Conservatively assuming a 2.1% increase in dose:

	Original Dose	New Dose	Change
LOCA*	0.4525 Rem	0.4620 Rem	0.01 Rem
CRDA	1.70 Rem	1.736 Rem	0.04 Rem

*Dose due to elevated release only; ground level release and direct shine doses are unaffected.

Although use of the correct sector would increase the CR elevated X/Qs slightly, the elevated release X/Qs used as input to the LOCA and CRDA calculations remain conservative. Fumigation was assumed for one-half hour during these accidents, and the PAVAN-calculated X/Qs were used for all time periods. Section 3.2.2 of RG 1.194 does not require consideration of fumigation and allows use of modified ARCON96-calculated X/Qs for periods greater than 2 hours, which would result in significantly smaller X/Qs.

Therefore, NMC does not propose to revise the submitted LOCA and CRDA calculations (CA-04-038 Rev 0 and CA-04-040 Rev 0). The Control Room X/Q calculation CA-04-037 Rev 2 will be revised to incorporate the proper sector so that the resulting X/Q is available for use as input to future dose assessments, but will not be resubmitted.

ENCLOSURE 5
Supporting Documents

1. MNGP Calculation CA-05-130, Rev 0, "Post LOCA Direct Dose to the Control Room from External Sources".
2. Drawing NF-36056-1, Rev H, EQUIP LOCATION - OFFICE BLDG PLAN AT ELEV 951'-0".
3. Drawing NF-36057, Rev L, EQUIPMENT LOCATION - REACTOR BLDG. PLANS AT EL.986'-6" & 1001'-2".
4. Drawing NF-36063, Rev A, EQUIPMENT LOCATION - REACTOR BLDG SECTION B-B.
5. MNGP Site Diagram.
6. Drawing NX-7831-197-1, Rev D, Reactor Vessel & Internals.